SAFIR2014 Annual Report 2012

Author: Kaisa Simola
Confidentiality: Public
The mission of National Nuclear Power Plant Safety Research programme 2011-2014 (SAFIR2014) 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. The SAFIR2014 Steering Group, responsible for steering and planning of the research programme, consists of representatives of the Finnish Nuclear Safety Authority (STUK), Ministry of Employment and the Economy (MEE), Technical Research Centre of Finland (VTT), Teollisuuden Voima Oyj (TVO), Fortum Power and Heat Oy (Fortum), Fennovoima Oy, Lappeenranta University of Technology (LUT), Aalto University (Aalto), Finnish Funding Agency for Technology and Innovation (Tekes) and Finnish Institute of Occupational Health (TTL). In September 2011 the Swedish Radiation Safety Authority (SSM) was also invited to join the SAFIR2014 Steering Group.

The realised volume of the SAFIR2014 programme in 2012 was 9.9 M€ and 71 person years. Main funding organisations in 2012 were State Waste Management Fund VYR with 5.6 M€ and VTT with 2.7 M€. The programme has been divided into nine research areas and in 2012 research was carried out in 42 projects.

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

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

VTT's contact address
VTT, PB 1000, FI-02044 VTT, Finland

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Preface

The mission of the National Nuclear Power Plant Safety Research Programme 2011-2014 (SAFIR2014), derived from the stipulations of the Finnish Nuclear Energy Act, is as follows: The objective of the SAFIR2014 research programme is to develop and maintain experimental research capability, as well as the safety assessment methods and nuclear safety expertise of Finnish nuclear power plants, in order that, should new matters related to nuclear safety arise, their significance can be assessed without delay.

The planning period 2011-2014 for the national research programme on nuclear power plant safety involves licensing processes for power plants that are in use or under construction, as well as overall safety evaluations related to licence terms. The construction works of the Olkiluoto 3 plant unit are proceeding, and the bidding and plant supplier phases of the Olkiluoto 4 and of the Hanhikivi 1 plant units are under way. These processes are reflected in many ways on national safety research.

Research on nuclear safety requires profound training and commitment. The research programme serves as an important environment providing long-term activity that is especially important now when the research community is facing a change of generation. During the planning period and in the following years, many of the experts who have taken part in construction and use of the currently operating plants are retiring. The licensing processes and the possibility of recruiting new personnel for safety-related research projects give an opportunity for experts from different generations to work together, facilitating knowledge transfer to the younger generation.

The SAFIR2014 research programme is divided into eight research areas: 1) Man, Organisation and Society, 2) Automation and Control Room, 3) Fuel Research and Reactor Analysis, 4) Thermal Hydraulics, 5) Severe Accidents, 6) Structural Safety of Reactor Circuits, 7) Construction Safety, and 8) Probabilistic Risk Analysis (PRA). Furthermore, projects can be focused on developing of the research infrastructure. Research projects of the programme are chosen on the basis of annual call for proposals.

In 2012 the realised volume of the SAFIR2014 programme was 9.9 M€ and 71 person years. Main funding organisations in 2012 were State Waste Management Fund VYR with 5.6 M€ and VTT with 2.7 M€. Research was carried out in 42 projects.

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

Espoo 22.4.2012

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1. Introduction

In December 2009, the Ministry of employment and the economy appointed a planning group to prepare a Framework Plan of the new national four-year research programme on nuclear safety for the period 2011-2014. The research programme is based on the Chapter 7a "Ensuring availability of expertise" of the Finnish Nuclear Energy Act.

The Framework Plan [1] contains a proposal for the general outline of the National Nuclear Power Plant Safety research programme, SAFIR2014. The programme covers essentially the themes of the SAFIR2010-programme (2007-2010) [2]. The plan has been made for the period 2011–2014, but it is based on safety challenges identified for a longer time span as well. Olkiluoto 3, the new nuclear power plant unit under construction and new decisions-in-principle made by Parliament on 1 July 2010 have also been taken into account in the plan. The construction of new power plant units will increase the need for experts in the field in Finland. At the same time, the retirement of the existing experts is continuing. These factors together will call for more education and training, in which active research activities play a key role. This situation also makes long-term safety research face a great challenge.

According to the Framework Plan, the research in the SAFIR2014 programme is organised by research areas. The eight research areas are:

1) Man, Organisation and Society
2) Automation and Control Room
3) Fuel Research and Reactor Analysis
4) Thermal Hydraulics
5) Severe Accidents
6) Structural Safety of Reactor Circuits
7) Construction Safety
8) Probabilistic Risk Analysis (PRA)

Furthermore, projects can be focused on developing of the research infrastructure (area 9).

The purpose of the Framework Plan is to provide information to the authors of project proposals on what topics are sought after and what are the main challenges and needs in each of the above-mentioned research areas. The Framework plan has been supplemented with research topics based on the needs for additional research identified after the Fukushima accident.

The public call for research proposals was announced at the beginning of October 2011. After the closure of the call, the SAFIR2014 steering group, taking into account the evaluations made by the reference groups, prepared a proposal for MEE regarding the set of projects to be funded in 2012. The funding decisions were made by the State Waste Management Fund (VYR) in March 2012. In 2012 the programme consisted of 42 research projects and the programme administration. 38 of the research projects are continuation to activities started in 2011, and four new projects were included in the programme.

The ‘VYR-funding’ is collected from the Finnish utilities Fortum Power& Heat Oy, Teollisuuden Voima Oy and Fennovoima Oy with respect of their MWth shares in Finnish nuclear power plants (units in operation, under construction and in planning phase according to the new decisions-in-principle). In addition to the VYR, also other key organisations operating in the area of nuclear safety are funding the programme.
The planned [3] and realised volumes of the SAFIR2014-programme in 2012 were 10.1 M€ and 9.9 M€ and 67 and 71 person years, respectively.

The SAFIR2014 Steering Group was nominated in September 2010. It consists of representatives of the Finnish Nuclear Safety Authority (STUK), Ministry of Employment and the Economy (MEE), Technical Research Centre of Finland (VTT), Teollisuuden Voima Oyj (TVO), Fortum Power and Heat Oy (Fortum), Fennovoima Oy, Lappeenranta University of Technology (LUT), Aalto University (Aalto), Finnish Funding Agency for Technology and Innovation (Tekes) and Finnish Institute of Occupational Health (TTL). In September 2011 the Swedish Radiation Safety Authority (SSM) was also invited to join the SAFIR2014 Steering Group.

Figure 1.1 Organisations represented in the SAFIR2014 steering group.

In the following, short summaries on the results of the individual projects (Chapter 2) and overall financial (Chapter 3) and administrative (Chapter 4) matters are given. Project publication lists are provided in Appendix 1, information on international co-operation in Appendix 2, list of Academic degrees obtained in Appendix 3, list of international travels in the projects in Appendix 4, and Appendix 5 contains list of the persons involved in the programme in Steering Group, Reference Groups and in the projects.
2. Main results of the research projects in 2012

The SAFIR2014 research programme is divided into nine areas:

1. Man, Organisation and Society
2. Automation and Control Room
3. Fuel Research and Reactor Analysis
4. Thermal Hydraulics
5. Severe Accidents
6. Structural Safety of Reactor Circuits
7. Construction Safety
8. Probabilistic Risk Analysis (PRA)
9. Development of Research Infrastructure

These areas are presented with more detailed descriptions of their research needs during the programme period 2011-2014 in the SAFIR2014 Framework Plan [1]. The research areas and research needs are based on the knowledge at the time of making the framework plan.

In 2012, the research was performed in altogether 42 research projects. The total volume of the programme was 9.9 M€ and 71 person years. The research projects in the various areas with their planned and realised volumes are given in Table 2.1.

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

Table 2.1. SAFIR2014 projects in 2012.

<table>
<thead>
<tr>
<th>Area</th>
<th>Project</th>
<th>Acronym</th>
<th>Organisation(s)</th>
<th>Planned funding (k€)</th>
<th>Real. funding (k€)</th>
<th>Planned volume (person months)</th>
<th>Real. volume (person months)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1.</td>
<td>Managing safety culture throughout the lifecycle of nuclear plants</td>
<td>MANSUCU</td>
<td>VTT</td>
<td>360</td>
<td>338.4</td>
<td>25</td>
<td>23</td>
</tr>
<tr>
<td></td>
<td>Sustainable and future oriented expertise</td>
<td>SAFEX2014</td>
<td>Aalto, TTL</td>
<td>101</td>
<td>74.2</td>
<td>9.5</td>
<td>6.8</td>
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<td>2.</td>
<td>Coverage and rationality of the software I&amp;C safety assurance</td>
<td>CORSICA</td>
<td>VTT, FiSMA Ry</td>
<td>181</td>
<td>181.1</td>
<td>18</td>
<td>14.9</td>
</tr>
<tr>
<td></td>
<td>Human-automation collaboration in incident and accident situations</td>
<td>HACAS</td>
<td>VTT</td>
<td>220</td>
<td>228.7</td>
<td>16</td>
<td>12.9</td>
</tr>
<tr>
<td></td>
<td>Safety evaluation and reliability analysis of nuclear automation</td>
<td>SARANA</td>
<td>VTT, Aalto</td>
<td>344.2</td>
<td>344.2</td>
<td>31.5</td>
<td>31.5</td>
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<tr>
<td></td>
<td>Safety requirements specification and management in nuclear power plants</td>
<td>SAREMAN</td>
<td>VTT, Aalto</td>
<td>190</td>
<td>155.6</td>
<td>14.7</td>
<td>13.7</td>
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Table 2.1. SAFIR2014 projects in 2012 (cont.).

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<tr>
<td>3.</td>
<td>Criticality safety and transport methods in reactor analysis</td>
<td>CRISTAL</td>
<td>VTT</td>
<td>228</td>
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<td></td>
<td>Three-dimensional reactor analyses</td>
<td>KOURA</td>
<td>VTT</td>
<td>288</td>
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<td></td>
<td>Development of Finnish Monte Carlo reactor physics code</td>
<td>KAARME</td>
<td>VTT</td>
<td>200</td>
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<td></td>
<td>Neutronics, nuclear fuel and burnup</td>
<td>NEPAL</td>
<td>Aalto</td>
<td>117</td>
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<tr>
<td></td>
<td>Extensive fuel modelling</td>
<td>PALAMA</td>
<td>VTT</td>
<td>335</td>
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<td></td>
<td>Radionuclide source term analysis</td>
<td>RASTA</td>
<td>VTT</td>
<td>61</td>
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<td>4.</td>
<td>Enhancement of safety evaluation tools</td>
<td>ESA</td>
<td>VTT</td>
<td>455</td>
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<td></td>
<td>Experimental studies on containment phenomena</td>
<td>EXCOP</td>
<td>LUT</td>
<td>266.4</td>
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<td></td>
<td>OpenFOAM CFD-solver for nuclear safety related flow simulations</td>
<td>NUFOAM</td>
<td>VTT, LUT, Aalto, Fortum</td>
<td>160</td>
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<tr>
<td></td>
<td>Numerical modelling of condensation pool</td>
<td>NUMPOOL</td>
<td>VTT</td>
<td>116</td>
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<td></td>
<td>Improvement of PACTEL facility simulation environment</td>
<td>PACSIM</td>
<td>LUT</td>
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<td>PWR PACTEL experiments</td>
<td>PAX</td>
<td>LUT</td>
<td>245</td>
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<td></td>
<td>Modelling of pressure transients in steam generators</td>
<td>SGEN</td>
<td>VTT</td>
<td>97</td>
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<tr>
<td></td>
<td>Uncertainty evaluation for best estimate analyses</td>
<td>UBEA</td>
<td>VTT</td>
<td>101</td>
</tr>
<tr>
<td>5.</td>
<td>Core debris coolability and environmental consequence analysis</td>
<td>COOLOCE-E</td>
<td>VTT</td>
<td>199</td>
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<td></td>
<td>Chemistry of fission products</td>
<td>FISKE</td>
<td>VTT</td>
<td>171</td>
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<tr>
<td></td>
<td>Thermal hydraulics of severe accidents</td>
<td>TERMOSEAN</td>
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<td>204</td>
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<tr>
<td></td>
<td>Transport and chemistry of fission products</td>
<td>TRAFI</td>
<td>VTT</td>
<td>317.5</td>
</tr>
<tr>
<td></td>
<td>Reactor vessel failures, vapour explosions and spent fuel pool accidents</td>
<td>VESPA</td>
<td>VTT</td>
<td>145</td>
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<table>
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<th></th>
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<th>Sponsor 2</th>
<th>Sponsor 3</th>
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<td>6.</td>
<td>Environmental influence on cracking susceptibility and ageing of nuclear materials</td>
<td>ENVIS</td>
<td>VTT</td>
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<td>Fracture assessment of reactor circuit</td>
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<td>VTT</td>
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<tr>
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<td>Monitoring of the structural integrity of materials and components in reactor circuit</td>
<td>MAKOMON</td>
<td>VTT</td>
<td>192</td>
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<td></td>
<td>RI-ISI analyses and inspection reliability of piping systems</td>
<td>RAIPSYS</td>
<td>VTT</td>
<td>152</td>
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<td>Advanced surveillance technique and embrittlement modelling</td>
<td>SURVIVE</td>
<td>VTT</td>
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<td>Water chemistry and plant operating reliability</td>
<td>WAPA</td>
<td>VTT</td>
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<td>Fatigue affected by residual stresses, environment and thermal fluctuations</td>
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<td>VTT</td>
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<td>Aging management of concrete structures in nuclear power plants</td>
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<td>VTT, Aalto</td>
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<td>Structural mechanics analyses of soft and hard impacts</td>
<td>SMASH</td>
<td>VTT</td>
<td>225</td>
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<td></td>
<td>Seismic safety of nuclear power plants. Targets for research and education</td>
<td>SESA</td>
<td>VTT, Aalto, Univ. Helsinki</td>
<td>142</td>
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<td>FMI</td>
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<td>Risk assessment of large fire loads</td>
<td>LARGO</td>
<td>VTT</td>
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<td>PRA development and application</td>
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<td>VTT</td>
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<td>9.</td>
<td>Enhancement of Lappeenranta instrumentation of nuclear safety experiments</td>
<td>ELAINE</td>
<td>LUT</td>
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<td>Renewal of hot cell infrastructure</td>
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<td>0.</td>
<td>Programme administration*</td>
<td>ADMIRE</td>
<td>VTT</td>
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*for period 1.1.2012-31.3.2013 with VAT 23% included
2.1 Man, organisation and society

In 2012 the research area "Man, organisation and society" consisted of two projects: Managing safety culture throughout the lifecycle of nuclear plants (MANSCU) and Sustainable and future oriented expertise (SAFEX2014).

2.1.1 Managing safety culture throughout the lifecycle of nuclear plants (MANSCU)

The generic objective of the MANSCU project (2011-2014) is to develop safety management approaches (such as rules and procedures, training, event investigation practices, safety culture evaluations and improvement programs, risk assessment and human performance programmes) in such a way that they would take into account the recent developments in safety science. MANSCU project focuses on three topics which are relevant for practitioners in the Nordic nuclear industry as well as supports the theoretical development of the safety culture model VTT has created. Safety management approaches should:

1) support the development of sufficient understanding and knowledge of nuclear safety and safety and risks as well as nuclear industry specific working practice demands.

2) take into account the needs of other contexts than the operating units. Especially we focus on creating knowledge on how safety culture should be understood in a) design activities and b) complex networks of subcontractors. The networks and supply chains may be multicultural and interdisciplinary.

3) support organisational alertness (mindfulness) to new risks or other unexpected conditions which are based on either technical or social phenomena. It also means avoidance of complacency and constant effort of continual improvement.

The MANSCU project is composed by three sub-projects:

- Safety culture and risk assessment in the design process (DESIGN). The goal of this subproject is to identify the organizational challenges associated with design and implementation activities and contribute toward better evaluation of the risks linked to new designs.

- Modelling methods for safe work processes in maintenance (MOREMO). The goal of this subproject is to find means to support flexible and adaptive maintenance work processes at the nuclear plants or constructions sites without sacrificing the safety of the decisions or the culture of complying with procedures.

- Theoretical reflections of safety culture model and safety management practises for future needs, (MODEL UPDATE). The goal of this subproject is to build up and maintain the expertise and theoretical models in the area of safety culture and human factors.

The research strategy of MANSCU project is that in addition to the theoretical work case studies are carried out in Finland and Sweden. The case studies are designed based on interests of the power companies and regulators and thus, they are not directly structured according to the MANSCU project generic goals. However, the purpose is that the practical case studies provide relevant material for gaining insights into the three above mentioned topics.
Specific goals in 2012

In 2012 the MANSCU project complements and analyses the data and experiences gained from the case studies concerning maintenance and design topics in 2011. While the case studies have their specific goals, all the experiences are put together and analysed from the viewpoint of what is the needed understanding for different employee groups as designers, maintenance workers and subcontractors. The special challenge of multinational employees is taken into account in the analysis.

The 2012 specific goals for the DESIGN subproject are:

- Identification, from the safety culture and resilience engineering perspective, of challenges and opportunities related to design activities in the nuclear industry based on the in-depth analysis of the interviews conducted in 2011 and 2012

- Preliminary model of human and organizational factors affecting the design process. The model incorporates the concepts of HFE, dimensions of safety culture and the necessary capabilities of resilience with empirical data on trade-offs, methods of assuring safety in design and practices of verification and validation

The 2012 specific goals for the MOREMO subproject are:

- Production of practical descriptions of resilient practices in different nuclear plants based on the combination and analysis of extensive data from four case studies

- Summary of lessons learned from the modelling methods

The 2012 specific goals for the MODEL UPDATE subproject are:

- Participation in IAEA’s work on safety culture

- Scientific publishing: 2 scientific articles and 3 conference papers

Deliverables in 2012

Figure 2.1.1.1. Cornerstones of safe activities (Oedewald, Pietikäinen & Reiman, 2011)
In the DESIGN subproject the following five major challenges related to design activities were identified:

1) Safety is not always the first and most important guiding value in the design process and commercial pressures may hinder safety

2) Understanding the context where the design will be utilized may be difficult for the designers and this may lead to dysfunctional designs

3) Safety philosophies may differ between different organizations

4) Coordinating activities may be difficult between organizations that work according to different logics and understandings

5) Distributing responsibilities and balancing roles between different stakeholders needs careful consideration

- In the DESIGN subproject a total of 64 opportunities for safety in design were found in the interviews and they have been categorised according to the three cornerstones of safety culture.

- Concerning HFE, in the DESIGN subproject it became evident that the maturity of organizations concerning HFE varies to the extent that it is not necessarily sure that organizations are addressing the same issues when they are discussing HFE related matters.

- In the MOREMO subproject resilient maintenance work practices have been identified and described.

- In the MOREMO subproject a methodology for evaluating the adjustments occurring in maintenance activities has been developed and applied.

- In the MOREMO subproject recommendations for improving the resilience of maintenance activities have been proposed

2.1.2 Sustainable and future oriented expertise (SAFEX2014)

In the near future, Finnish nuclear power industry is facing several challenges in the area of knowledge and competence management and expertise development. Reasons for these challenges include start-up of new power plants, global labor markets, and “baby boomers” retirement peak. Two preceding SAFIR funded research projects by the applicants (Disseminating Tacit Knowledge and Expertise in Nuclear Power Plants, Diamond 2004-2006 and Expert work in safety critical environment, SafeExpertNet, so called SAFEX 2007-2010) identified individual and organizational factors related to expert work and development of expertise in nuclear power industry.

The objective of the project is to provide information on the management and development of human resources in organizations in nuclear energy industry.

Specific goals in 2012

One specific goal in 2012 includes assessment and follow-up of well-being and expertise development in nuclear industry organizations. Participation on this survey gives organizations an opportunity to evaluate how different kinds of human resource practices are experienced by employees (e.g. stress and well-being, opportunities to develop expertise, supervisory work, etc.). One theoretical model behind the survey is presented in figure 2.1.2.1.
In the same survey the interorganizational collaboration between nuclear energy experts is evaluated. Respondents are asked to evaluate e.g. the frequency, objectives, and obstacles of interorganizational collaboration. Figure 2.1.2.2 shows that experts in the industry value objectives of the collaboration very high, but actual collaboration is not realized alike.

Figure 2.1.2.1. Job demands and resources model (Schaufeli, & Bakker 2004, Hakanen 2009).

Figure 2.1.2.2. Respondents view on inter-organizational collaboration.
Deliverables in 2012

Project delivered six different reports to participating organizations. In five organizations also a vocal presentation was organized. Reports are based on SAFEX-survey, which has been carried out also in years 2008 and 2010. Participating organizations can have thus longitudinal information on how the measured variables (e.g. job resources/demands, efforts to develop expertise, supervisory work, inter-organizational collaboration) have changed or developed over time.

Project delivered a prototype of an assessment method for evaluating hr-practices in nuclear energy industry. The prototype is based on the literature review, interviews among nuclear energy industry experts, and two workshops with reference group members. The assessment method will be piloted in year 2013 and improved version will be delivered to the organizations in 2014.

2.2 Automation and control room

In 2012 the research area "Automation and control room" consisted of four projects: Coverage and rationality of the software I&C safety assurance (CORSICA), Human-automation collaboration in incident and accident situations (HACAS), Safety evaluation and reliability analysis of nuclear automation (SARANA) and Safety requirements specification and management in nuclear power plants (SAREMAN).

2.2.1 Coverage and rationality of the software I&C safety assurance (CORSICA)

There is a general need to perform qualification of I&C safety systems as effectively as possible, keeping still the necessary formalism and accuracy of the qualification. Early notification of potential problems is the most effective and proactive way to perform qualification. Good knowledge of nuclear standards, professional application of sophisticated methods and good ability to adopt novel technologies is essential for qualification.

In the previous SAFIR2010 program we developed approaches to qualify and certify software intensive I&C systems for nuclear power plants. This work is continuing in the current SAFIR2014 program, extending previous results for general evaluation for example in following topics:

- adequacy and relevance of process capability assessment in technical product evaluation,
- coverage and rationality of required development and assurance methods,
- certification and evaluation issues in using new technologies, for example field-programmable gate arrays (FPGA),
- use of new standards in technical safety evaluation of nuclear I&C systems

Specific goals in 2012

- Developing integrated Nuclear SPICE to assess development processes as part of certification or qualification of safety critical I&C systems and software.
• To produce a mechanism to convert safety-critical process assessment outcomes into a software reliability value.

• Comparing the review process of a reactor trip system in respect of specific acceptance criteria by comparing US regulations with the Finnish regulations.

• Development of a review technique for nuclear domain design documentation, and a pilot case study using the developed technique.

• Survey new technologies that have been or are likely to be introduced into NPPs in the near future.

![Diagram](image)

**Figure 2.2.1.1. The integration layers of standards and criteria in Nuclear SPICE**

**Deliverables in 2012**

• The continuous development of the Nuclear SPICE framework. The development produces a novel solution to a specific business need i.e. the need to assess processes that are used to develop software with utmost safety requirements. By adding content and criteria from generic safety standards and from nuclear standards in Nuclear SPICE, a holistic method can be done to assess the process capability and compliance. This idea is presented in Figure 2.2.1.1. The full scale Nuclear SPICE model is the baseline, but more specific models are also needed. Such an assessment model and assessment process was developed for pre-qualification phase of nuclear I&C systems. Main target is software pre-qualification in safety categories A and B+C.

• Specifying tentative set of process quality attributes for process assessment in safety domain. First, software reliability was studied from process point-of-view; software reliability quantification is a controversial issue. Instead, a process assessment framework to evaluate safety characteristics of software development processes was developed based on a new Process Quality concept. The basic set includes attributes that meet the elementary requirements for trustworthy software development. The aim is that risks related to achievement of safety goals can be evaluated with process assessment.

• Comparing U.S. NRC reactor trip software review process to the Finnish regulatory requirements. Nowadays, several instrument and control (I&C) systems have been accepted by the U.S. Nuclear Regulatory Commission (NRC) and offered to Finnish
nuclear markets. Identifying the difference between the NRC and STUK regulatory requirements makes the approval of their systems easier. We have identified the main differences between the review process of the NRC and the Finnish regulatory requirements when a typical reactor trip software system is reviewed.

- Development of software reading techniques in the nuclear domain. We have applied the generic ideas of the perspective-based reading technique to the review of nuclear domain conceptual design plans. A conceptual design plan is a high-level design document that is used as input when the detailed design of the system is made. We first analysed the development life-cycle and the use of the conceptual design plans by interviews. Based on this analysis we developed a review technique that consists of five different review perspectives. Separate review instructions (i.e. a scenario) were written for each perspective: an automation designer scenario, a control room designer scenario, an electrical designer scenario, a safety designer scenario, and a regulator scenario. The main novelty of the developed technique is that the reviewer can try to simulate the future work phases, and try to anticipate what kind of practical issues may not have been considered in the conceptual design phase.

- Introductory report on multi-core processing from the NPP I&C perspective. A literature review was written focusing on the use of multi-core processors from the reliability point of view. The report gives an overview of the technology and differences from single-core processors with descriptions of some of the methods used for improving computing performance. The impact on reliability is considered along with potential ways to use the technology.

- Implementation of an I&C case study using Field-programmable gate array technology. A case study was used to take a closer look at application design, implementation and verification. A logic circuit called the Stepwise Shutdown System (SWS) was implemented. The case study gave a good overview of the FPGA design process from the requirements to the actual programming and testing of the device. Although some comprehensive testing methods were developed, the analysis of the output-file data is still rudimentary. The design was successfully tested against the requirements specification, via both simulation and hardware testing, which was the main purpose of the testing in the case study.

2.2.2 Human-automation collaboration in incident and accident situations (HACAS)

The project focuses on studying how digital automation and control room (CR) upgrades affect resilient performance of CR personnel, and how humans and automation systems collaborate to accomplish safety and production goals of nuclear power plants (NPPs). Specifically, the project aims at gathering knowledge of control room operators’ procedure usage and human-system interaction in accident management, with the aim to identify the operators’ capabilities of supporting resilience of the entire system. The project also contributes to identifying Human Factors Engineering (HFE) considerations that are important in the selection of CR modernization strategies. In more focussed studies the use of interactive large screen displays in process control will be investigated. An important further task is to develop a new type of requirement-based validation approach for stepwise implementation of CR changes. In this connection, new ways of implementing risk assessment methods in the validation will also be investigated. The final issue under investigation is the effect of the operators’ automation awareness on operator behaviour in
incident situations. Methods and tools will be developed for the evaluation of automation awareness and competence.

As the output of the research, practical methods and guidelines for the design of operating procedures and HSIs supporting operator practices, and methods and guidelines for the training of their application and usage are developed. Tools and guidelines are developed to support different stakeholders in the accomplishment of HFE activities during different stages of the plant life cycle. Knowledge on factors affecting automation awareness and automation skills in plants based on digital automation and HSI technologies, and methods for their development are provided. As results of this work, a detailed Concept of Operations (ConOps) for emergency situations can be developed.

### Specific goals in 2012

In 2012, the research in the project has focussed on six main areas: First, we have studied the effect of routines of procedure usage on the management of severe accidents, the parallel use of EOPs and safety HSIs in accident management, and the procedure design process from the HFE perspective. Second, we have studied the effect of digital I&C systems on operator practices in accident situations. Third, we have prepared a review of the HFE implications of the level of the HSI modernization and of the migration strategy. Fourth, we have identified challenges in the design of interactive large screen displays (LSDs) for the simulator environments and conducted a simulator test on the interactive large screen usage. Fifth, P. Laitio has prepared her Master’s thesis on operators’ automation awareness and on the effect of automation complexity on automation awareness and automation skills. Sixth, we have designed an Apros-based simulation demonstrator for testing automation awareness in experimental settings (see Figure 2.2.2.1).

### Deliverables in 2012

- Resilience engineering –based approach for the analysis of work practices has been outlined. There are five main steps in the resilience-based analysis approach: analysis of the work system, modeling the domain and control demands, modeling situational task demands, analysis of actual behaviour, and identification of generic tool functions. The systems usability analysis tools have been applied for empirical evaluations of NPP CRs. In these studies it has been found that performance-based criteria deliver important information of the tool’s instrumental capabilities, and the practice-based and user experience-based measures are particularly valuable in informing of the tool capabilities with regard to the psychological and communicative functions.

- The usage of emergency operating procedures in a simulated accident scenario has been analysed in a detailed fashion. The variance between the operator crews concerns the functions of information usage, interpretation of the situation, dealing with automation, decision making, communication, and leadership. The different habits of action for each function are described and the habits are graded according to their capability of producing system level resilience in the operating activity. According to the results, procedure usage should be actively trained and in the training attention should be paid, not only to following the procedure but also to the way of utilising the procedure in a manner which does not detach procederalized tasks from the environment.
Figure 2.2.2.1. Examples of the simulator displays.

- It has been studied operator practices in accident management in a digitalized control room from the perspective of resilience engineering and automation awareness. Based on the literature review, it was found that resilience and automation awareness provide parallel viewpoints on digitalisation of nuclear processes and user interface, and the existence of positive effect for resilience usually means a positive effect on automation awareness and vice versa. Additionally, the effect of some given solution on resilience and automation awareness is not straightforward but the solution can encompass both pros and cons.

- A review of the HFE implications of the level of the HSI modernization and of the migration strategy has been prepared. The review is based on literature data and on interviews of managers, designers and operating personnel in Finland and on discussions with two representatives of the Swedish Radiation Safety Authority (SSM). The results have been analysed and synthesized to obtain an overall view of the prospects and limitations of different modernization levels and strategies. According to the literature review, a fully modernized digital CR provides a plenty of benefits over the old analogue one, but the risks involved in the modernization project itself are high that it has to be carefully think about whether they are worth the benefits. Guidelines that are based on findings from several modernization projects) are provided to help in developing a strategy for a successful implementation of a CR modernization.
• A descriptive model of the innovative features of the Virtual panel concept within the System Usability framework was refined, and a paper has been prepared on the topic. Overall, the Virtual panel development has been an invaluable learning experience for the whole multidisciplinary design team, and many innovative solutions but also defensible compromises has needed to be made in order to realize the new digital panel system. A simulator study has been conducted at the Fortum Loviisa training simulator in which the aim has been to test the usability of individual features of the panel system, and the systems usability of the hybrid HSI complex of the turbine side. The results of the test will be analysed and reported in 2013.

• Paula Laitio has prepared her Master’s thesis on automation awareness. The term automation awareness is introduced, and automation awareness is defined as a part of situation awareness. The development of automation awareness is considered to be a continuous process that consists of perceiving the current status of the automation, comprehending the status and its meaning to the system behaviour and estimating its future status and its meaning. The question of what is sufficient automation awareness and how to examine automation awareness is addressed. In addition, literature-based guidelines for successful human-automation collaboration are presented. Based on the theoretical work, a preliminary version of a questionnaire-based evaluation tool has been developed for the measurement of automation awareness and automation skills.

• The development of an Apros-based simulator environment that will be used in the experimental testing of automation awareness has continued, and the ProcSee visualizations have been linked to Apros.

2.2.3 Safety evaluation and reliability analysis of nuclear automation (SARANA)

The general objective of the SARANA project is to develop methods and tools for safety and reliability analysis of digital systems and to utilize them in practical case studies. The project covers various topics and aims to build bridge between risk analysis and automation experts. SARANA provides guidelines to analyse and model digital systems in Probabilistic Risk Assessment (PRA) context, brings together deterministic and probabilistic analyses for safety assessment of plant designs, develops the reliability analysis tool YADRAT, develops an iterative and automatic algorithm for modular model checking of large systems, and develops more efficient methods for systematic model checking of asynchronous systems. This summary report introduces the on-going work and some of the intermediate results of the project.

Specific goals in 2012

One of the objectives of the project is to develop practical guidelines for analysis and modelling of digital systems in probabilistic reliability analysis (PRA) for nuclear power plants. In 2012, the target was to prepare draft guidelines which to be submitted to a wider audience for commenting. Fictive digital protection system example will be used to study the effect of different modelling approaches, e.g. related to handling of fail-safe principles.

The framework based on the standard IEC61508 developed in 2011 will be further developed with focus on the assessment of software common cause failure (CCF). Tentatively the scope is limited to application software and in the assessment of the probability of CCF in a) same application and same platform, b) same application, different platforms, c) different applications, same platform and d) different application, different platforms. The assessment
process can be thus divided into two tasks: 1) what is the failure probability of application software and 2) what is the degree of diversity between the applications and the platforms.

The Dynamic Flowgraph Methodology (DFM) is an approach for reliability analysis of digital instrumentation and control systems. The YADRAT tool developed at VTT is based on interpreting the DFM model as a Binary Decision Diagram (BDD). The objective is to develop a qualified tool for the reliability analysis of dynamic systems. The aim in 2012 is to study ways to improve the scalability of YADRAT and to continue the work on risk importance measures and common cause failures in the domain of DFM modelling. Additionally, the aim is to make a benchmark study between two DFM approaches YADRAT and DYMONDA with an American company ASCA Inc. who is the developer of DYMONDA.

In 2011, we created a model checking algorithm that iteratively searches for a composition of modules that at the same time is computationally manageable, and covers enough modules to prove the fulfillment of temporal properties in the original model. In 2012, the objectives are to continue the theoretical development of the algorithm and also implement the theoretical improvements.

A new modelling approach was earlier started for the UPPAAL model checker that allowed subsystems, each operating with their own clocks, to be modelled accurately. Doing so, the goal was to find fault scenarios that are difficult to reproduce otherwise. This work will be continued in 2012 and extended also to the NuSMV model checker input language. In addition to developing new modelling techniques, the aim is also to investigate the possibility of model checking tool additions tailored for the above described globally asynchronous, locally synchronous systems (GALS) analysis.

An important question in employing model checking is the reliability of the tool: If the model checker says the system is correct, can we trust the result or is the model checker itself possibly buggy? This concern is tried to be alleviated creating an alternate model checking tool chain to analyze NuSMV models that do not share source code with the NuSMV codebase.

One more objective is to create links between the different methods used for modeling and analyzing nuclear I&C in different projects in the SAFIR2014, such as SARANA, SAREMAN, CORSICA, HACAS, and MANSCU. The aim is to organize a workshop between the projects and to get an overall picture about the projects and on which levels of abstraction they operate.

**Deliverables in 2012**

- Best practice guidelines on failure modes taxonomy for reliability assessment of digital I&C systems for PSA have been developed.

- A representative fictive digital protection system example has been developed to demonstrate the applicability of the above mentioned taxonomy in the transparent manner.

- An approach to model independent post-release evaluation of safety-related software using Bayesian belief net, with the specific aim to assess the probability of failure on demand (pfd) of the software has been developed.
• A method for the assessment of software common cause failures in the reactor protection system has been developed.

• Connections and links between the different methods used for modeling and analyzing nuclear I&C in different projects in the SAFIR2014, such as SARANA, SAREMAN, CORSICA, HACAS, and MANSUCU have been clarified and a joint workshop in the area has been organized.

• A benchmark study between two DFM approaches YADRAT and DYMONDA has been started.

• The theory for the risk importance measure calculation and common cause failure modelling has been developed for the YADRAT tool.

• The iterative model checking algorithm has been revised and simplified in order to focus more on proving properties quickly. In our technique the system is divided into modules. Abstractions of the model are created by replacing a subset of the modules with non-deterministic interface modules. Iterative abstraction refinement is used to obtain a computationally manageable subset of the modules that is sufficient for proving the given formal specification. Our refinement procedure is based on traversing the dependency graph of the modules, and first finding some feasible refinement to the abstraction. The feasibility of the refinement can be checked rather quickly using a bounded model checking algorithm. After finding a suitable refinement, we use heuristics to minimize that refinement. In order to improve the performance of our technique we utilize two model checking algorithms in parallel. The overall algorithm is depicted in Figure 2.2.3.1 below.

![Figure 2.2.3.1. Overall algorithm for verifying large modular systems by model checking.](image)

• Modelling techniques to model asynchronous systems with Uppaal and NuSMV model checkers have been studied and a preliminary NuSMV-based implementation has been created for model checking asynchronous circuits with timing components.
• The NuSMV independent SMV model checking tool chain has been created based on using the standardized AIGER input format. The tooling accepts a large subset of the SMV language.

2.2.4 Safety requirements specification and management in nuclear power plants (SAREMAN)

Unambiguous requirements are a prerequisite not only for successful implementation but also for demonstrating the safety of nuclear power plant systems. The aim of the SAREMAN research project is to develop good practices for requirements specification and management in nuclear power plants. SAREMAN is planned to be a four-year project. From a consistent, domain-specific terminology the project proceeds to requirement modelling and documentation methods, working practices and finally to utilization of information technology in requirements engineering (figure below). In order to limit the scope SAREMAN focuses on I&C systems. This highlights the multi-disciplinary nature of requirements engineering involving, for example, safety engineering, process engineering and control room operations. The principles of requirements engineering are, however, common to several application areas. So, the results of the project are widely applicable to the design and licensing of nuclear power plants.

![Figure 2.2.4.1. Overall work plan for the four-year period.](image)

During the first project year 2011, the focus was on terminology and unambiguous, readable requirements. Recommendations were collected for the key documents, the high-level Concept of Operations (ConOps) and the technical System Requirements Specification (SyRS). In 2012 the research focused on three areas: 1) traceability of design information; 2) guidance for requirements specifications; and 3) regulatory compliance and role of requirements in safety demonstrations. The availability of researchers was limited. Therefore, some planned deliverables were had to be dropped or postponed to 2013. Correspondingly, only 82% of the planned costs were realised. The research activities during 2012 (14.7 person-months in total) are described in more detail below.

Specific goals in 2012

1. Conceptual model of requirements traceability: The aim was to complete and refine the working report from 2011 titled “Conceptual model for safety requirements specification and management in nuclear power plants”. In particular, traceability and configuration aspects were elaborated.
2. **Documentation practices**: The goal was to collect recommendations for the content and structure of high-quality requirement specifications. The task concentrated on two main topics: structured natural language templates and recommendations on information contents and document structures. The work started in 2011 with structured natural language requirements and document templates suffered from the lack of resources. However, the task was able to produce two conference papers and an introductory guide to requirements engineering, primarily intended for control engineers.

3. **Cross-disciplinary requirements definition**: The purpose of this task was to study multi-disciplinary working practices in the development of a Concept of Operations (ConOps) and I&C requirements. To ensure their applicability of ConOps, this work was linked to the current documentation practices, in particular to the development of structured safety demonstrations, e.g., Safety Analysis Reports (SAR) and Safety Cases.

4. **Practices for ensuring regulatory compliance throughout the life-cycle of a system**: The challenge addressed in this task was how regulatory requirements should be tracked throughout the life-cycle of a system. After a preparatory phase, this task continued in a Master’s thesis work started in October. The focus has been on understanding requirements from regulations viewpoint. The well-known V-model has been studied as a general reference model. The Master’s thesis will be finalised in 2013.

![Life-cycle activities organised as a V-model.](image)

**Figure 2.2.4.2. Life-cycle activities organised as a V-model.**

**Deliverables in 2012**

- **Conceptual model for safety requirements specification and management in nuclear power plants, version 2**: The goal of this working report is to provide clear terminology as a foundation for systematic design. One of its claims is that requirements cannot be discussed in isolation from other engineering activities and system descriptions. The second version developed in 2012 contains additions concerning the traceability of design information (separate report below). Moreover, the description of life-cycle activities has been refined (figure above).

- **On modelling traceability in requirements engineering**: A literature review prepared as a background material for the extended conceptual model (above). It showed that traceability is necessary for safety demonstrations and required by the standards and regulations. No single definition or a way to implement traceability can be found. Advanced traceability is rarely applied in the industry due to its complexity and costs.
Traceability can’t be developed in isolation, but the structure of design artefacts, as well as the design and management processes should be improved simultaneously.

- **Concept of operations (ConOps) in the design of nuclear power plant instrumentation & control systems, version 2**: The idea of Concept of Operations (ConOps), included in many guidelines and standards, could be used as a basis for multi-disciplinary requirements elicitation and early safety assessment. The aim of this working report is to interpret the general ideas for the design of I&C systems in nuclear power plants. The second version discusses the applicability of the Assurance/Safety Case approach for demonstrating safety of NPP systems. A Systems Usability Case applied to a control room system is described as an example.

- **Role of requirements in safety demonstrations**: A literature review collected as a background for the second release of the ConOps report above. Here, the main observation is that in the current practices, Safety Analysis Reports seem to describe the design solutions more carefully than the requirements behind them. Moreover, a systematic argumentation to show how the requirements will be or have been satisfied is not always present. So, the claim-argument-evidence chain in the heart of a Assurance/Safety Case could be used to improve the clarity of safety demonstrations.

- **A control engineer’s introduction to requirements engineering**: First version of a document that summarises the cumulated research results as an informal guide primarily written for control engineers in the nuclear domain. This booklet tells what requirements are and what their role is in the design process. It gives hints for digging out the needs of various user groups, transforming them into well-formed requirements and managing requirements during the design. Important standards, regulatory guidelines and further readings are shortly described. There are many internal and external links, so this guide is most useful in electronic form.

- **Practices for ensuring system compliance with regulatory requirements throughout a design cycle**: A Master’s thesis is in progress in the SAREMAN project. The objectives of the thesis are to better understand information produced during the design cycle in practice, and in theory. On the one hand, the study analyses currently produced information expressed in various documents. On the other hand, the study analyses existing reference guidelines such as IEC and ISO standards. The common frame of reference is the V-model of development life cycle.

### 2.3 Fuel research and reactor analysis

In 2012 the research area "Fuel Research and Reactor Analysis" consisted of six projects: Criticality safety and transport methods in reactor analysis (CRISTAL), Three-dimensional reactor analyses (KOURA), Development of Finnish Monte Carlo reactor physics code (KÄÄRME), Neutronics, nuclear fuel and burnup (NEPAL), Extensive fuel modelling (PALAMA) and Radionuclide source term analysis (RASTA).

#### 2.3.1 Criticality safety and transport methods in reactor analysis (CRISTAL)

Reactor physics is a base element in nuclear power safety. The CRISTAL project is an essential platform for educating new experts in the field of reactor physics. A wide range of codes and applications are dealt with. The expertise in the field has to be increased in order to
ensure appropriate and adequate safety analyses. In addition to a knowledgeable staff, good enough coverage is needed on the code side. The project also tackles this issue. The codes have to be kept up to date as well as the understanding of them. In addition, their validation has to be brought to an appropriate level in accordance with international standards.

Specific goals in 2012

Specific goals in 2012 include a lot of work with the object in improving the expertise of the researchers. The training of one person into inventory calculation started in 2011 was to be continued. The knowledge of CASMO-code was to be increased through preparation of a cross section calculation package for BWRs.

The UAM benchmark was continued in 2012. The methodology of the generalised perturbation theory was to be extended to other relevant assembly parameters than two-group homogenised cross sections. These parameters include e.g. assembly discontinuity factors and pin powers.

In criticality safety the main emphasis was in creating a criticality safety validation package. This effort is a long-lasting and tedious work. The package is based on the International Handbook of Evaluated Criticality Safety Benchmark Experiments. It is important that the criticality validation is brought to an appropriate level in accordance with international standards.

![Figure 2.3.1.1. Geometry of the bottom of part of the LR-0 model for the validation package. The figure shows the bottom of the control cluster on the left and the nest and the base of the fuel assembly on the right.](image)

Deliverables in 2012

- A script was written to take care of various calculation steps of cross section preparation for BWRs. This script, BWRXS.pl, uses given CASMO-inputs as source and runs CAXMAN and CRFIT creating their inputs on the fly. As a result it produces cross sections in the HEXBU-format. As a first application of this package, a cross-section set was prepared for the BWR stability benchmark studied in KOURA.
• HEXBU-3D was successfully transferred to and compiled on the Linux system.

• New experience on decay calculations and 3D-discrete-ordinates method was obtained through calculation of the VVER-440 CB6 spent fuel final disposal benchmark. A major obstruction was encountered as the code used in the work did not converge in all the cases due to round-off errors in the single-precision code. Some unsuccessful attempts were made to obtain a double precision version of TORT, claimed to solve the convergence problems. The result depends on the US export control issues.

• The development of the generalised perturbation theory (GPT) methods for sensitivity and uncertainty analysis in CASMO-4 has continued. The goal is to extend the methodology to cover all the assembly parameters that are passed to core simulators.

• The building of a criticality safety validation package for the VVER-440 reactors for MCNP was continued. The initial set of modelled benchmarks was increased by 25 new cases from the LR-0 reactor. This set includes fuel assemblies with enrichment of 3.6 % and 4.4 %. It covers assembly pitches from 18 to 19.25 cm. It also contains cases with and without absorbing elements around the assemblies. The package now comprises 48 modelled critical benchmark experiments. The number of cases has grown large enough that the analysis of the cases becomes soon possible.

• The inter-laboratory “Round Robin” dosimetry exercise involving precise activity measurements was participated in. 10 laboratories participated in this exercise. The samples to be measured were activated metallic foils of Ag, Co, Fe, Ti and Nb. The results of the exercise are to be collected and summarised in early 2013 and officially reported at the 15th ISRD in May 2014.

• A “Master” cross section library in the SAND-II format (640 energy groups up to 20 MeV) has been prepared. This library is mainly based on the newly-released IRDFF library (in ENDF-6 format) with a few additional reactions of interest for dosimetry and isotope production, mostly taken from ENDF-B/VII.

### 2.3.2 Three-dimensional reactor analyses (KOURA)

The goal of the project is to have a truly independent transient calculation system, which can be utilized by the safety authority and other end-users for safety analyses. To achieve this, VTT’s reactor dynamics codes must be constantly developed in order to be on the same level as other codes used for similar purposes internationally. One of the main objectives in this project is supplementing the code system with three-dimensional thermal hydraulics modelling.

In addition to the development work itself, it is essential that the new models are validated against measurements and the results of other codes. Moreover, the expertise in the reactor dynamics field has to be increased.

**Specific goals in 2012**

The project has two main research areas. The objective of the first area is to develop methodology for 3D two-phase thermal-hydraulic modelling of the nuclear reactor core and to increase knowledge of the thermal hydraulic phenomena in the nuclear reactor core and pressure vessel. In practise, this means further development of VTT’s in-house PORFLO code
and modelling of the reactor pressure vessel with PORFLO. PORFLO is a 3D solution code that utilizes the concept of porous medium and it is designed to analyse thermal hydraulic phenomena in multiphase flow problems related to nuclear power plant safety analyses.

The objective of the second area is to improve modelling methods for reactor dynamic phenomena, improve models in dynamic reactor analysis codes TRAB-3D and HEXTTRAN, as well as test and validate these methods against plant measurements and code-to-code comparisons. Much of this work can be done as international co-operation in the form of calculating benchmark problems organized by e.g. OECD/NEA and AER. It is also important to increase the understanding of different phenomena in reactor core and in the whole plant.

**Deliverables in 2012**

- The PORFLO code has been further developed and work has been reported. Parallelization based on domain decomposition and MPI calls has been finalized. The mesh and several solutions can now be written to the same cgns-file, which enables the use of third party post-processing software for visualization purposes.

- PORFLO’s capability to handle different element types has been tested. The calculation of cell-centered gradients, a cornerstone of the collocated SIMPLE algorithm, has been improved and some of the previous convergence problems encountered with geometrically unstructured meshes have been resolved. Symmetrical boundary condition has been implemented, which reduces the amount of calculation cells needed whenever symmetry can be utilized. The k-ε turbulence model has been implemented and tested. Also the first version of material properties of water and steam (steam tables), based on IAPWS-IF97 polynomial functions, has been written, but found to be time consuming.

- Unstructured hexahedral mesh for an EPR pressure vessel was created and preliminary solution of the turbulent single-phase flow field was computed with PORFLO. For a RPV quarter, a hybrid mesh of 250 000 (hexahedral and tetrahedral) cells was created, Figure 2.3.2.1. RPV simulations were further enhanced by including the heat transfer modelling. Results of the simulations have been reported.

- Previous applications of PORFLO code have been introduced in the conference paper presented in the ICONE-20 conference.

- VTT participates to the OECD/NEA O2 benchmark, in which a stability event due to a feedwater transient at the Oskarshamn-2 BWR plant will be modelled. In 2012 the TRAB-3D model for the Oskarshamn-2 nuclear reactor has been constructed and the model has been tested in stationary conditions. The geometry of the hydraulic circuit model is shown in Figure 2.3.2.2. A report on the model has been written.

- Pin-power reconstruction method and computer model have been thoroughly studied and some minor modifications have been done. An output option providing the boundary conditions for pin power reconstruction has been added to the current version of TRAB-3D and tested. Work has been reported.
Figure 2.3.2.1. Visualizations of the unstructured mesh for an EPR pressure vessel.

Figure 2.3.2.2. Geometry of TRAB3D circuit model for Oskarshamn-2.
First validation calculations of the internally coupled TRAB3D-SMABRE code with a BWR pressurization transient against plant data have been carried out, with very promising results, Figure 2.3.2.3. Deficiencies in the pellet heat transfer modelling and some hydraulic submodels have been identified, needing further work. Work has been reported.

Figure 2.3.2.3. Measured and calculated reactor power and system pressure with TRAB3D and TRAB3D-SMABRE in Olkiluoto 1 overpressurization transient 1985.

- Stand-alone and coupled TRAB3D versions have been merged into a new version TRAB3D 4.0. Unnecessary source code that has been used during the development of internal coupling e.g. for debugging has been removed.

- Material and knowledge on generating reactor physics input data for reactor dynamics codes have been searched from files and documents and reviewed, archived for the project staff. Due to generation changes and partly inadequate documenting of silent knowledge some of these connections have been tedious to find or understand, and some methods will have to be re-created.

- Instructions on conservative methods of carrying out accident analyses with 3D reactor dynamics codes, available in HEXTRAN since the early 90s, have been collected and the methods have been taken into use in a PWR model in TRAB-3D.
• The conference paper of the initialization of the fuel rod properties with the fuel performance code ENIGMA for reactor dynamic codes has been presented in the 22nd AER symposium.

• In addition to the above mentioned ICONE-20 conference and AER symposium, the project included participation in the AER working group D and meeting of the AER scientific council. The project included also participation in the meetings of the OECD/NEA Working Party on Scientific Issues of Reactor Systems WPRS and the meetings of its two expert groups: first on Reactor Physics and Advanced Nuclear Systems EGRPANS, second on Uncertainty Analysis in Modelling EGUAM. The Scientific Workshop on Reactor Dynamics and Safety, which was focused on stability analysis was attended as well.

2.3.3 Development of a Finnish Monte Carlo reactor physics code (KÄÄRME)

The Serpent Monte Carlo code has been developed at VTT since 2004, and mainly funded from the EMERALD and TOPAS reactor physics projects in the previous SAFIR programmes. In SAFIR2014, the development of Serpent is gathered under its own, well-specified project, with two major goals:

1) To maintain the publicly available version of the Serpent code (Serpent 1), distributed by two international data centres (OECD/NEA and RSICC)

2) To develop a new version of the source code (Serpent 2), with improved features and entirely new capabilities

Serpent 1 user community includes more than 200 registered users in 88 universities and research organizations in 27 countries around the world. Serpent 2 entered a beta-testing phase in January 2012, and the test group has grown to 90 users with previous experience in Serpent 1. The project has close collaboration with the PALAMA, NEPAL and CRISTAL projects in SAFIR2014.

Specific goals in 2012

Re-writing of the Serpent source code started in September 2010, and most of the efforts in the KÄÄRME project are devoted to the development of Serpent 2. The first half of the four-year project has focused on parallelization and the fundamental transport and burnup capabilities. A beta-version was made available to registered users in January 2012, and most of the capabilities in Serpent 1 are now available in the new version.

Deliverables in 2012

• Improvements in memory management and parallelization were completed during 2012. The Serpent 2 code now has the capability to handle tens or hundreds of thousands of depletion zones without limitations regarding parallelization. Different optimization modes were developed for small and large burnup calculation problems, and the methodology was presented at the PHYSOR-2012 conference.

• Research on spatial oscillations and stability of the burnup algorithms has been continued in collaboration with the NEPAL project.
• Several methods for producing homogenized multi-group constants for deterministic reactor simulator calculations were adopted from Serpent 1. Most of the capabilities in Serpent 1 are now also available in Serpent 2.

• Doctoral thesis on numerical methods for burnup calculations, in particular the development of the Chebyshev Rational Approximation Method (CRAM) for solving the full system of depletion equations has been completed.

Figure 2.3.3.1. Simulated neutron tracks in a VVER-440 fuel lattice. The track plotter is a new feature in Serpent 2, developed for visualizing source distributions and the neutron random walk.

• Development of the on-the-fly explicit temperature treatment routine has been continued. This method, together with a new capability to model non-uniform density distributions, forms the basis of multi-physics coupling in Serpent 2.

• Development of a universal multi-physics interface has been started. In the framework of the KÄÄRME project this interface will be used for coupling Serpent 2 to fuel performance codes. The same interface can be applied to thermal hydraulics coupling, which supports the NUMPS multi-physics project funded by Academy of Finland.

• Development of the internal temperature feedback routine has been continued, and moved from NEPAL to KÄÄRME.

• International collaboration and interaction with the Serpent user community included participating in an international reactor physics conference PHYSOR-2012 and the ANS
Winter Meeting, seminars and workshops at foreign universities and research organizations, maintaining a website and a discussion forum1 and direct communication with users via e-mail. The Second International Serpent User Group Meeting was organized with the Polytechnic University of Madrid in Madrid, Spain, 19-21, 2012. The three-day event brought together 40 Serpent users from 16 organizations around the world.

- Three scientific papers have been published or accepted for publication and three more papers submitted. Five conference papers were presented and five more will be presented in May 2013.

2.3.4 Neutronics, nuclear fuel and burnup (NEPAL)

The Fission and Radiation Physics Group at Aalto University School of Science concentrates on developing calculation methods for reactor physics, modeling basic physical and chemical phenomena in nuclear fuel, and researching new fuel cycles and next generation nuclear reactors. The activities seamlessly combine education and research of nuclear engineering. The essential field of know-how of the group covers physics-based analyses and numerical computation.

The behavior of high-burnup fuel in a quasi-stationary situation mainly depends on the mechanical strength of the cladding and its ability to transfer heat to the coolant. The characteristics of fuel pellets are described on the basis of empirical data. Reliable modeling outside of the normal operating parameters necessitates thorough understanding of the phenomena and their modeling in a mesoscopic scale. Behavior of a porous, chemically complex medium in a radiation field is a challenge to model. Besides mechanical strength the composition of nuclear fuel is important to know for radioactivity analysis methods and disposal of spent nuclear fuel.

The general objectives of the NEPAL project are the following:

- Development of methodology in Monte Carlo code systems: code comparisons (e.g., Serpent, FLUKA, MCNP) and their new applications (Triga, GEN4, Th-assemblies).
- Development of computational efficiency and accuracy in various subroutines, evaluation of selected nuclear constants.
- Coupling of Monte Carlo neutronics codes with temperature distribution of fuel pellet (Doppler effect).
- Accurate calculation of nuclide concentrations, especially concerning rare but potentially important nuclides.
- Mesoscopic description and modeling of nuclear fuel.
- Education of experts: YTERA doctoral programme, strengthening expertise on nuclear fuel, knowledge on core calculation codes and their applications in special cases (e.g., Th-assemblies in LWR core).
- Preparedness for new international projects, e.g., JHR.

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Specific goals in 2012

We model the accurate composition of uranium and as a special case Th-based LWR-fuel at high burnup (above 60 MWd/kgHM), developing burnup algorithms and comparing their accuracy and performance. In 2012, the emphasis was on spatial oscillations in Monte Carlo calculations. Accurate burnup calculations aim at finding rare but potentially problematic nuclides. These could comprise strong absorbers or other reactor-physically important nuclides. Additionally, it could be important to know accurate concentrations of nuclides that are important for spent fuel disposal or nuclear safeguards.

The exact internal temperature distribution of a fuel pellet is modeled. For this purpose, methods are developed for temperature-dependent neutronics calculation. The results are coupled to a multi-physics model that can be utilized for modeling heat transfer, swelling, cracking and pellet-cladding interaction (PCI). In 2012, a Master’s thesis was written on this theme, more specifically on development and implementation of temperature feedback in Serpent.

We construct a mesoscopic simulation model of the thermal creep failure of fuel pellets. The model includes damage accumulation from radiation-induced fission gas buildup, and the behavior of the gases themselves. The buildup induces microcracking which couples back to the gas dynamics. Development of the novel model started in late 2011 and the first results from a functional model were expected in 2012.

Deliverables in 2012

- Existing Monte Carlo burnup codes suffer from xenon driven spatial oscillations large geometries. Since the accurate solutions to the simulation models for many of these geometries oscillate, simply reducing step lengths, or solving the model better, would not work in a general case. When using predictor-corrector methods these oscillations can be difficult to detect as they can happen between the predictor and corrector, instead of subsequent steps, which makes the usually collected results look stable.

Forcing xenon and flux to a mutual equilibrium offers a simple solution to these oscillations. Since the equilibrium calculation can be integrated to normal Monte Carlo neutronics, it has only a minor effect on running times and can be used with any burnup calculation algorithm. Furthermore, at least the equilibrium iteration algorithm of Serpent can significantly improve flux convergence in large geometries with high dominance ratio.
Oscillations driven by fuel burnup still arise if too long steps are used, but unlike xenon oscillations, these only occur with long steps, allowing calculations to be performed with reasonable step lengths. This is a major improvement, especially in cases where stable solutions simply could not be obtained without the equilibrium method. It is important to remember that while the equilibrium xenon treatment allows stable solutions to be obtained for any model, the equilibrium levels are only as accurate as the model they have been calculated for. (Isotalo et al., 2013)

• Our studies of temperature-related effects on neutronics were continued with a Master’s thesis aiming to design and implement a temperature solver routine for the Serpent code. The routine is used in conjunction with the on-the-fly Doppler processor already found in Serpent 2 to provide high resolution temperature discretization with minimal memory usage. This leads, however, to a significant increase in the computation time needed for simulations.

The new routine addresses the radial heat transfer in fresh 2D pin-geometries. The consideration of 3D-problems and addressing the effect of the burnup on the problem are, however, deemed to be realistic directions of future development for the routine. Likewise, the temperature distribution could also be solved for non-pin-like geometries as long as certain simplifications such as symmetry considerations can be applied.

A thorough literature review was made for choosing default internal material property correlations for the routine and an integral comparison was performed against the fuel code Femaxi-6. The correspondence between the results was found to be good.

Two assembly level calculations verified the capability of the code to handle problems while monitoring various fuel pins of different compositions. The simulations also gave quantitative results on the differences between using the temperature feedback system and a single homogeneous volume-averaged temperature for the fuel part of the bundle combined with Doppler pre-processing. (Valtavirta, 2012)

![Figure 2.3.4.2. The rod-wise volume averaged temperatures at the top-left quadrant of a 17x17 PWR fuel assembly. The red squares contain Gd-doped UO2 rods and the black squares instrumentation and guide tubes. (Valtavirta, 2012)](image-url)
The microstructure of nuclear fuel changes significantly during reactor operation. As fuel undergoes fission, heat and fission products are released. Since oxide fuels have low heat conductivity, the center of a fuel pellet becomes hotter than the outer region. An uneven temperature distribution causes radial fragmentation during early stages of a fuel pellet’s lifetime. Thermal creep due to thermal stress leads to further microcracking over time.

In addition to thermomechanical effects, fission products play an important role in the fuel’s microstructural evolution. Stable gases, such as several isotopes of xenon and krypton, make up a significant portion of all fission products. Gas atoms diffuse inside the pellet, and eventually form bubbles both within grains (intragranular) and grain faces (intergranular). The formation of bubbles increases the fuel’s porosity, causing swelling and lowering the fuel’s thermal conductivity. Gas released from the fuel mixes with the fill-gas (pressurized helium) in the pellet-cladding gap, decreasing thermal conductivity between the pellet surface and cladding. The behavior of fission gases is an important safety issue, since they can potentially cause the fuel assembly to overheat, or lead to pellet-cladding interactions due to swelling. Economically it makes sense to use fuel up to higher burnups, but the amount of fission products increases with burnup as well.

Gas diffusion and thermal creep are coupled together. Both gas bubbles and microcracks make the fuel more porous over time. This can affect the rate of gas release due to gas diffusion occurring through pore networks in addition to atomistic diffusion. There is evidence of fission gas release via coalescing gas bubbles along the grain boundaries. The goal of this subproject is to combine porosity formation from thermal creep fracture and gas accumulation to the diffusion of generated gas within a fuel pellet.

We are developing a computational model for simulating the microstructural evolution of nuclear fuel. The model includes damage accumulation from thermal creep deformation and from fission gas buildup within the pellet. Damage accumulation is linked with increasing porosity of the fuel, as microcracks and gas bubbles are formed. Diffusion of fission gases is simulated from the viewpoint of percolation theory: gas flows through interlinked pores, and can only reach the surface of the pellet through continuous pore pathways.

Currently, the basic components of the model have been implemented into a computer code and tested. Preliminary results demonstrate how gas diffusion is affected by local porosity. When the average porosity is lower than the percolation threshold, only a small fraction of gas generated near the edges of the system is released. Most of the gas is trapped in the inner parts, from where there are no open pathways to the surface. Once the average porosity is increased over the critical threshold, most of the generated gas is released.

The next step is to modify some parts of the model. Simultaneous time evolution of porosity and the gas inventory should be implemented, so that porosity is directly linked to the amount of gas. A realistic value for the critical porosity and other simulation parameters should also be determined. It will be interesting to see how different rates of damage accumulation from creep and radiation damage, together with a pore closure mechanism, will affect the dynamics of the gas in the system. (Ovaska, 2013)

2.3.5 Extensive fuel modelling (PALAMA)

The general goal of PALAMA project is to develop and maintain competence and tools required for independent nuclear fuel behaviour assessment in both normal operation and
accident conditions. The upcoming update to the regulatory guides creates a need to better understand, describe and model phenomena related to increasing burnup and the statistical nature of the fuel rod behaviour. Investigation into so-called design extension conditions requires fuel behaviour studies broadened towards areas such as thermal hydraulics, reactor dynamics and severe accidents. The increasing volume of Finnish nuclear power production induces new challenges as the load following operation may be required and its effects to the fuel performance must be understood. The fuel performance codes need to be systematically validated, and the creation of a framework for such a purpose is one of the goals of the project. The tradition of strong bilateral cooperation with foreign and international organizations and international collaboration in the form of participation in benchmarking programmes, working groups and conferences is upheld.

**Specific Goals in 2012**

Specific goals for 2012 include continuation of the validation project for the fuel behaviour code ENIGMA initiated in 2011, update of the statistical versions of the steady state fuel behaviour codes and strong participation in the international benchmark programmes. The international co-operation such as fuel behaviour part of VTT - Halden Reactor Project in-kind work, participation in working groups OECD/CSNI WGFS and ETSON SAG, as well as the following of CABRI and JHIP progress is done under this project.

**Deliverables in 2012**

- A review on phenomena relevant to fuel performance under load following operations was made.
- Models describing BWR fuel behavior during RIA scenarios were created for SCANAIR as a part of researcher visit to IRSN which was funded in part by PALAMA and in part by VTT.

![Graph showing Yield stress and Ultimate tensile stress for fresh and irradiated fuel](image)

**Figure 2.3.5.1. YS0.2 and UTS of fresh and irradiated Zry-2. Different correlations were tried in order to find the best match to the measurements.**

- A Master’s thesis was initiated (to be completed in early 2013) on cladding oxide layer formation.
- A statistical script for FRAPCON was created, and assumptions of linearity and additivity of uncertainties was studied.
- FINIX (Fuel behaviour model for multiphysics applications) was created. It is a simple fuel rod model intended for source-code level implementation to other codes.
• New boundary conditions implemented into the FRAPTRAN-GENFLO coupled fuel behaviour / thermal hydraulics code in 2011 were verified as a part of PALAMA 2012 work.

• The validation database for SPACE (Simulation Performance Analysis Code for ENIGMA) for simulation validation work was updated and User’s Manual written for the system.

• Halden IFA-650 LOCA experiments 5, 6 and 7 were analysed in co-operation with Quantum Technologies (Sweden).

• WGFS RIA benchmark calculations were performed with SCANAIR code and the results reported in the benchmark conference.

• The 6th workshop of OECD/NEA Benchmark for Uncertainty Analysis in Best-Estimate Modelling for Design, Operation and Safety Analysis of LWRs was attended. VTT has been providing input for the formulation of Phase II fuel behaviour exercise.

• Behaviour of Gd-doped fuel was investigated with ENIGMA fuel behaviour code, and new recommendations on simulating the dimensional changes of Gd-fuel were formulated.

• Methodology to take account the transient stress in creep models which is able to duplicate the results of Halden tests has been developed and implemented in ENIGMA. Results of the IFA-610 overpressure series have been further reviewed and a proposal for driving mechanism has been suggested.

• Several international conferences were participated in. Information event on international fuel behaviour projects was organized in June 19th.

2.3.6 Radionuclide source term analysis (RASTA)

In the work performed in one-year project RASTA the tools of reactor physics, fuel behaviour and severe accident analysis are investigated in order to find synergy between the strengths of different disciplines. Individual investigations are described also to serve as a reference of the use of the tools and the viewpoint utilized by different disciplines. This is important as it is well recognized that the first obstacle to fruitful co-operation is the one’s inability to understand the mental framework and assumptions of the experts of other fields.

Long-term activity inventory calculations have been performed extensively with code ORIGEN-S both internationally and in the VTT. The calculated inventories from ORIGEN can be used as input for various accident analysis tools which make assumptions regarding distribution and behaviour of different radionuclides. The development of Monte Carlo based reactor physics calculation tool Serpent has given a tool to do these calculations with more spatial precision, thus enabling investigations in the scale of the fuel rod.

Fuel behaviour codes traditionally track only fission products that may affect rod integrity, in other words stable noble gases and in some cases iodine. The focus of the analysis has been the failure of the rod, not the subsequent release. Especially in the final repository assessments the similarity and differences between the behaviour of noble gases and the radionuclides have been investigated, but so far very little actual modelling has been performed.
The research of severe accident in-vessel phenomena have recently focused at VTT on core degradation and coolability. This approach has not enabled detailed analysis on radionuclide release. The integral codes only assume, that the radionuclide source term is initiated by diffusion from fuel grains, which does not take pre-existing radionuclide deposits or radically modified fuel structure into account. This model is assumed to give incorrect estimations in the case of slowly progressing accidents.

One of the focus areas of the work done in RASTA is the effect of increasing burnup, subsequent formation of high burnup structure (HBS) and its effect on the radionuclide release. As the Finnish utilities have gained permission to increase discharge burnup from the old 45 MWd/kgU, the highest power rods are well within the burnup region where HBS begins to form, and its effect should be determined.

In the course of the work done in RASTA several separate tasks were performed:

- Activity inventories were calculated for a PWR rod for a burn-up of 51.22 MWd/kgU with point-depletion code ORIGEN-S. The calculation also included a cooling (decay) period of 10,000 years. For reference, the same case was calculated with Monte Carlo code Serpent. The main difference between the methods is that as a point-depletion code ORIGEN-S does not provide information on the distribution of nuclides within a rod, while Serpent model used divided the rod into 15 axial rings. Effect of different cross section libraries was studied with Serpent using four different libraries.

- Amount of readily released nuclides during accident or in repository were studied by literature survey as well as by implementing the ANSI/ANS-5.4 recommended models into VTT’s fuel be-haviour code ENIGMA. The limitations of these models were discussed as well as the effect of high burnup structure which is not taken into account by the models that are based solely on conventional fission gas release models. Effect of environment to the actual release was briefly discussed.

- State of the art of the severe accident analysis regarding radionuclide release was examined. The various radioactive nuclides were listed and their behaviour in case of severe accident was detailed. Example code runs with ASTEC were performed, with an attempt to simulate also the effect of the high burnup structure.

When discussing the effect of burnup to the amount of releasable radionuclides the most important effect comes from the increasing inventory. While some of the nuclides reach a stable level early on during the irradiation, others keep accumulating during the whole irradiation period. The relative fraction of short-lived radionuclides which are released after rod failure in handling-type accidents can be assumed to be fairly stable, as long as the fuel temperature has been that of the usual operation. The fractional amount of longer-lived isotopes that should be in the free volume is assumed to be linearly related to traditional noble gas release, but even that varies over a batch of rods. Traditionally high burnup structure is not taken into account (Table 2.3.6.1), but depending on the accident scenario or repository lifetime assumptions it could be justified in some cases, as the potential increase in fast released fraction is large.

The amount of inventory residing in HBS is not straightforward to determine from e.g. burnup alone. While it probably would be possible to determine rim inventories to specific nuclides parametrically, the ability of Monte Carlo neutronics simulations to provide spatial nuclide distributions provide sufficient information for such analysis. This is assuming HBS inventory is relevant, as it appears to be in loss of coolant accident scenarios. For repository analysis the
jury is still out, and the early effect may be vanishingly small for full-blown severe accident scenario.
Table 2.3.6.1. Assumptions made in various models on nuclide release.

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<th>Half-life</th>
<th>Grain release model</th>
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ANSI-5.4 standard provides guidelines for radionuclide release from the fuel. The standard has several limits to its applicability, especially the limit to sudden temperature increase as well as demand that no oxidation takes place. Temperatures that are high for reactor conditions are discussed in the RASTA final report, especially regarding HBS effect, and severe accident conditions have a part of the report of their own. The oxidation of the fuel is taken into account in ASTEC fuel models. Taking those models as a starting point would be logical should model development be initiated for scenarios where fuel oxidation is important, such as operation with defective fuel. Also, ANSI-5.4 is based on the measurements of Kr-85m, and other radioisotopes are connected to that with some correcting factors. Therefore it should be noted that while the different cross-section libraries that are commonly used produce very similar results for isotope compositions, the very Kr-85 yield is different depending on the library used.

The literature on accident analysis contradicts the assumptions made for repository analysis on some issues, such as the rate of diffusion of several key radionuclides (Table 2.3.6.1). As these are related to chemical characteristics of the fission products and not methodology, they should be ultimately unified according to the best understanding.

As the severe accident analysis codes focus on the situations where the fuel melts, the early release from (relatively) intact fuel is hardly a focus for them. As it is, the radionuclide release from the fuel is not very rigorously modelled in e.g. MELCOR. ASTEC has its own model for fuel grains, which works but is known not to predict all the dynamics of radionuclide release. HBS is supposed to be modelled as small separate grains, which has implications to early interpretation of the HBS behaviour. However, as the HBS appears to release gases at high temperature transients, the model is somewhat justified, even if not necessarily mechanistic. As it is the HBS portion of the fuel could be modelled with some difficulty, unfortunately it would appear that the ASTEC model for grain release is either not functioning or just does not affect the end result much. Also the gap inventory, which in theory could also be used to model the HBS as clad failure might happen only after HBS burst, is added to the released mass fraction at the beginning of the simulation regardless of the temperature in ASTEC.
Deliverables in 2012

- Project, as previously described, was fully performed during 2012, and the project report (Nieminen Anna, Räty Antti, Tulkki Ville, 2013. "Radionuclide source term analysis", VTT Re-search Report VTT-R-00647-13) was the sole deliverable.

2.4 Thermal hydraulics

In 2012 the research area "Thermal Hydraulics" consisted of eight projects: Enhancement of safety evaluation tools (ESA), Experimental studies on containment phenomena (EXCOP), OpenFOAM CFD-solver for nuclear safety related flow simulations (NUFOAM), Numerical modelling of condensation pool (NUMPOOL), Improvement of PACTEL facility simulation environment (PACSIM), PWR PACTEL experiments (PAX), Modelling of pressure transients in steam generators (SGEN) and Uncertainty evaluation for best estimate analyses (UBEA).

2.4.1 Enhancement of safety evaluation tools (ESA)

The main objectives of the project were to develop and validate calculation methods for safety evaluation of nuclear power plants. Both thermal hydraulic system analysis codes and CFD calculations were used in the analysis and their usability was studied and enhanced. APROS and TRACE codes were validated with both separate effect and integral tests concentrating on the phenomena important for the new plant concepts offered for the Olkiluoto 4 and the Hanhikivi 1 projects. An important objective was also to train new thermal hydraulic code users and educate young experts.

Validation of system analysis codes

The thermal hydraulic system analysis codes APROS and TRACE were validated with experimental data from Lappeenranta University, EU and OECD research programs.

Heat transfer degradation in horizontal steam generator due to non-condensable gas was studied in PACTEL experiments NCg-1, NCg-3, NCG2-04 and NCG2-05. The tests were modelled with APROS to validate the heat transfer models of the code (Figure 2.4.1.4). This validation also serves modelling of condensers in the new power plant concepts since the thermal hydraulic conditions with non-condensable gases are similar. Stepwise injection of non-condensable gas decreased heat transfer in the steam generator and consequently system pressure increased after each injection.
TRACE and APROS codes were validated by calculating integral experiments simulating intermediate break loss-of-coolant (IBLOCA) accidents OECD/ROSA2 Test 2 and Test 7. The experiments were conducted in LSTF test facility in Japan (Figure 2.4.1.5), which is a large, 1/48 volumetric scaled, model of a PWR. The TRACE model utilized 3D vessel component, which usability was separately studied by calculating the OECD/ROCOM mixing Test 2.2.

Containment thermal hydraulics

A Generic Containment Benchmark exercise, organised by SARNET2 (EU Severe Accident Research Network) was modelled with APROS Containment. The benchmark basic nodalization was developed for a German pressurized water reactor (PWR) with 1300 MW<sub>el</sub> provided by GRS (Figure 2.4.1.6). All participants tried to build-up an identical nodalization concept/simulation model. In the last step (run-2), the Generic Containment was applied as a basis for testing different passive auto-catalytic recombiner (PAR) modelling approaches. A PAR system consisting of 57 AREVA type PARs, was modelled.
Figure 2.4.1.6. Nodalization for the German pressurized water reactor containment.

Containment modelling tools were further validated by participating in the German THAI benchmark. The experiment TH24 was intended to investigate the dissolution of a steam layer subject to natural convection in the atmosphere of the THAI test vessel. The experiment was calculated with both the Fluent CFD code and the APROS lumped parameter (LP) containment code. The main component of the THAI facility is a cylindrical steel vessel of 9.2 m height and 3.2 m diameter, with a total volume of 60 m³ (Figure 2.4.1.7).

Figure 2.4.1.7. THAI test facility (left), APROS (middle) and Fluent (right) nodalizations.

To enhance modelling capabilities a building condenser experiment (GEKO) was modelled with APROS Containment. The experiment simulated the building condenser in the Kerena plant concept with finned condenser tubes (Figure 2.4.1.8). The effect of the fins on heat transfer was modelled by defining an efficiency coefficient.
Deliverables in 2012

- APROS was validated with non-condensable gas PACTEL experiments NCg-1, NCg-3, NCG2-04 and NCG2-05.

- APROS and TRACE codes were validated with the integral experiments of OECD/ROSA2 research program studying cold leg intermediate brake loss-of-coolant (IBLOCA) accidents.

- TRACE 3D vessel models were evaluated by modeling ROCOM downcomer mixing test 2.2.

- The AREVA correlation for passive auto-catalytic recombiners (PAR) was implemented as an alternative model in the APROS Containment.

- APROS containment modeling was verified with SARNET generic containment benchmark in which passive auto-catalytic recombiners (PAR) were included in the test case.

- APROS containment applicability was extended to building condensers by calculating the GEKO building condenser experiments

- Fluent and APROS containment modeling approaches of gas stratification were tested in THAI TH24 benchmark exercise.

2.4.2 Experimental studies on containment phenomena (EXCOP)

The main goal of the EXCOP project is to gather an extensive experimental database on condensation dynamics, heat transfer and structural loads in a suppression pool environment to be used for testing and developing computational methods used for nuclear safety analysis, such as Fluent, NEPTUNE_CFD, TransAT, GOTHIC, APROS and TRACE. The behaviour
at the blowdown pipe outlet during air/steam discharge still needs to be investigated experimentally in more detail to improve simulation models. The PPOOLEX test facility at Lappeenranta University of Technology (LUT), including models of the Boiling Water Reactor (BWR) drywell and wetwell compartments and withstanding prototypical system pressure, has been used in the experiments. To achieve to above mentioned goals sophisticated measuring solutions i.e. a Particle Image Velocimetry (PIV) system and a modern high speed camera have been installed to the PPOOLEX facility in 2011-2012. The applicability of measurement results for the validation of CFD, lumped parameter and structural analysis codes has increased considerably as a result of these improvements in instrumentation. Networking among international research organizations has been enhanced via participation in the NORTHNET framework and NKS/ENPOOL project.

Specific goals in 2012

Specific goals in 2012 included two experiment series with the PPOOLEX test facility, constructing a Passive Containment Cooling System (PCCS) model and simulation of previous experiments with NEPTUNE_CFD and TransAT codes.

In case of small steam flow rates, thermal stratification could develop above the blowdown pipe exit and significantly impede the pressure suppression capacity of the condensation pool. Experimental studies have shown that once steam flow rate increases significantly, momentum introduced by the steam injection and/or periodic expansion and collapse of large steam bubbles due to direct contact condensation (DCC) can destroy stratified layers and lead to mixing of the pool water. Accurate and computationally efficient prediction of the pool thermal-hydraulics with thermal stratification, mixing, and transition between them, presents a computational challenge. KTH is developing and implementing the Effective Heat Source (EHS) and Effective Momentum Source (EMS) models in GOTHIC code. To provide necessary data for the development of the models, a series of experiments in the PPOOLEX facility at LUT was carried out in 2012. A fine resolution both in space and time for detection of the dynamics of free water surface in the blowdown pipe was essential to get an accurate estimation of the frequency and amplitude of oscillations for the assessment of the effective momentum. This was achieved with an extensive net of measurements added into the blowdown pipe before the experiments.

During the small steam flow rate period temperatures below the blowdown pipe outlet remained constant while increasing heat-up occurred towards the pool surface layers indicating strong thermal stratification (Figure 2.4.2.1). As the experiments proceeded to the mixing phase the steam mass flow rate was rapidly increased to get the steam/water-interface moving up and down inside the blowdown pipe. Typically the prevailing condensation modes were steam condensation within vents or blowdown pipes and chugging. Depending on the used steam flow rate and the degree of thermal stratification total mixing of the pool volume was achieved in 150–500 seconds. If the pool water temperature was let to increase above 50 °C the water inventory began to stratify again even though the steam flow rate was not decreased. The experiments revealed that during the mixing phase the steam/water-interface oscillated up and down inside the blowdown pipe with amplitude of 29–999 mm and frequency of 0.6–1.82 Hz depending on the used steam flow rate and prevailing pool water temperature.
Figure 2.4.2.1. Development of vertical temperature profile in pool water in a PPOOLEX stratification/mixing experiment.

A PIV measurement system was installed to the PPOOLEX facility in 2011-2012 for capturing the details of DCC related phenomena. The idea is to get CFD grade measurement data for verification/validation of numerical models. Severe problems with the PC controlling the PIV system delayed the beginning of the DCC experiments planned for 2012. The experiments were started at the end of the year with some water and steam injection tests. Initial results show that the steam discharge environment of the PPOOLEX facility is very challenging for obtaining reliable PIV measurements. The use of laser-induced fluorescence (LIF) seeding particles will be tested next. With those the disturbing impact of small gas/steam bubbles could possibly be filtered out from the raw images.

The PCCS is a system that passively removes heat from the containment to the liquid pool surrounding the PCCS heat exchangers. In addition to cooling it also has an important role in mitigating the offsite dose by retention of a fission product release in the containment. A scaled PCCS condenser model of the type used in ABWRs was designed and constructed in 2012 (Figure 2.4.2.2). PPOOLEX will act as a host facility in the forthcoming PCCS experiments. The main objective of the experiments is to gain data for the validation of severe accident code MELCOR. The systems for aerosol feeding and for measuring the deposition of aerosols in the condenser tubes will be designed in cooperation with VTT in 2013.
Figure 2.4.2.2. PCCS model next to the PPOOLEX host facility.

A pattern recognition analysis of bubble sizes and their frequency in the POOLEX STB-28 chugging mode test was carried out in 2012. Similar qualitative behavior between the test and the corresponding CFD simulations (NEPTUNE_CFD and TransAT) was observed. Also a brief simulation study comparing saturated and superheated steam blowdowns was done in order to expand the modelling of BWR suppression pools to the supercritical water reactor (SCWR) context. The results indicate that superheating has a predictably small - but not negligible - effect on the condensation rate. The effect of vapor side heat transfer seems to be very small both in the superheated and saturated steam cases. Regardless of the relatively small difference in the condensation rates of saturated and superheated steam, a major difference in qualitative behavior between the cases was seen. Probably due to the smaller density of superheated steam, the volume of superheated steam collapsed more rapidly than the volume of saturated steam although the condensation rates predicted opposite.

**Deliverables in 2012**

- A series steam stratification/mixing experiments was carried out in the PPOOLEX facility. An extensive net of measurements was added into the blowdown pipe for the detection of the dynamics of free water surface in the pipe. Total mixing of the pool volume was achieved in all experiments by increasing the steam discharge rate so that steam condensation within vents or blowdown pipes and chugging were the dominating condensation modes. The experiments provided necessary data for the development of the EMS and EHS models to be implemented in GOTHIC code by KTH.

- A scaled PCCS condenser model of the type used in ABWRs was designed, constructed and connected to the PPOOLEX facility.

- Simulations of POOLEX experiments STB-28 and STB-31 were continued with NEPTUNE_CFD and TransAT codes. Furthermore, a pattern recognition analysis of bubble sizes and their frequency in the POOLEX STB-28 chugging mode test was done. A brief simulation study comparing saturated and superheated steam blowdowns was conducted in order to expand the modelling of BWR suppression pools to the SCWR context.
Networking among international research organizations was enhanced via participation in the NORTHNET framework and NKS/ENPOOL project. NORTHNET Roadmap 3 meetings were participated in Stockholm and Lappeenranta and the status of condensation pool research in participating organisations (LUT, VTT and KTH) was discussed. A combined ENPOOL-NKS funding application for 2013 by LUT, VTT and KTH was written.

2.4.3 OpenFOAM CFD-solver for nuclear safety related flow simulations (NuFOAM)

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 policies prevent effective utilization of parallel computer resources. Use of open source software in the nuclear safety analysis would increase its transparency as more parties will have an access to the simulation tools. More effective utilization of parallel computing would allow a use of more complex physical models and detailed computational grids. This would improve the accuracy and reliability of modeling and increase the number of situations where CFD methods can be utilized.

The main goal of the NuFoam project is to validate an open source CFD-software OpenFOAM as a tool for nuclear safety related simulations. A national network of OpenFOAM users performing nuclear safety CFD analysis is further developed, and connections with the international OpenFOAM users and developers are formed. International co-operation will be utilized in order to minimize the national development work.

In single-phase flow simulations, the aim is simulation of flow and a heat transfer in a complex geometry, especially in a fuel assembly and its head parts. As a result an improved understanding of the coolant mixing in a fuel rod bundle is achieved. In two-phase simulations, an OpenFOAM based solver is developed for modeling sub-cooled nucleate boiling in fuel rod bundles. The results can be used in verifying safety issues when increasing the burn up of the fuel.

Specific goals in 2012

Single-phase flow and heat transfer are important topics in nuclear applications. A fuel rod bundle is still among the ‘grand challenges’ of CFD. Huge computational grids are required by the relatively large size of the fuel rod bundle, which has small and complex geometrical details. The spacer grids are difficult to model accurately and grids affect the heat transfer and pressure losses in the fuel rod bundle. A whole-length bundle is beyond the limits of the practical computer resources available, although a few cases could be simulated using hundreds of CPU cores. Heat transfer is less studied aspect of the flow in comparison to friction.

Recently, LES-based methods including detached eddy simulation (DES) have become more popular in CFD research, and the aim of this sub-project is to continue the studies started in 2010. In 2011, the main purpose was to verify the RANS and DES approaches. Based on the simulations performed in 2011 DES simulations were continued in 2012 with a revised single sub-channel model. These simulations were done by applying the RANS, LES and DES turbulence modelling approaches with OpenFOAM.

The OECD/NEA/CSNI/WGAMA blind CFD benchmarking activity was participated, which was based on the rod bundle experiment MATIS-H. The calculation of one of the benchmark
cases was submitted and the results were selected as the best approximation with the $k-\varepsilon$ turbulence model.

Heat transfer solvers for two-phase flows are being developed and validated. The main application is subcooled nucleate boiling in fuel rod bundles, but the two-phase solver has also been extended to simulations of direct contact condensation (DCC) and the goal is to extend the range of applications further. The energy equations were included in an OpenFOAM two-phase solver at VTT during year 2011. Co-operation with the Royal Institute of Technology (RIT) was started by exchanging two-phase solvers developed at RIT and in the NuFoam project. The first version of the subcooled boiling model was implemented in the two-phase solver by the end of year 2011.

In 2012, the development and testing of the subcooled nucleate boiling model was continued. A new default turbulence model, $k-\omega$ SST, and support for non-uniform bubble diameter was implemented. Different models for the bubble diameter were also implemented and the numerical robustness of the solver was improved.

DEBORA 5 and 6 experiments, where air bubbles flow upwards in a vertical pipe filled with water, were recalculated with different turbulence model formulations. The $k-\varepsilon$ turbulence model results were better than earlier, but the level of turbulence was still significantly under-predicted. The $k-\omega$ SST model predicted more reasonable turbulence levels. The OECD/NRC PSBT sub-channel benchmark 1.2211 experiments were simulated utilizing a non-uniform bubble diameter model and a selection of different interfacial force and turbulence models. The results match well with simulation results published in literature.

During 2012, numerical calculations of condensing flow in POOLEX experiment STB-31 were performed by using version v0.3 of the two-phase solver. In those calculations, the steam flow rate was very low so that the steam-water interface was kept steady close to the blowdown pipe outlet. Numerical modeling of direct-contact condensation (DCC) was carried out by using the direct contact condensation models implemented previously to NEPTUNE_CFD code, for instance, Hughes-Duffey based DCC model and DCC model of Lakehal et al.

![Figure 2.4.3.1. Instantaneous velocity distributions in the single sub-channel DES simulation. SST-SAS turbulence model. Periodic boundary conditions.](image-url)
Deliverables in 2012

- Reports on single-phase RANS simulations.
- Reports on modelling subcooled nucleate boiling: physical models, implementation and validation.
- Report on modelling direct-contact condensation in two-phase flows.

2.4.4 Numerical modelling of condensation pool (NUMPOOL)

The objective of the project is to develop two-phase and Fluid-Structure Interaction (FSI) calculation methods that can be used for modelling of a pressure suppression pool of a BWR. In particular, the direct-contact condensation occurring in the pool during a hypothetical large-break loss-of-coolant is modelled. The loads on the wall structures are calculated with Computational Fluid Dynamics (CFD) and the structural behaviour is calculated with Finite Element Method (FEM). Coupled CFD-FEM calculations are performed to analyse the effect of the loads on wall structures. The numerical calculations are validated by comparing the results to the experiments performed with the PPOOLEX facility.

The NUMPOOL project is part of the ENPOOL project, which is partly funded by NKS. The ENPOOL project consist of the combined effort done at LUT, VTT and KTH to implement the ideas outlined in the NORTHNET Roadmap 3 on experiments and modelling of pressure suppression pools. The work done in the ENPOOL project is reported to the NORTHNET Roadmap 3 Reference Group.

In 2012, the chugging phase in the pressure suppression pool was studied in the NUMPOOL project. Rapid condensation of almost pure vapour was modelled with CFD calculations. Simulations of PPOOLEX experiments were performed to determine the pool behaviour in the chugging phase. The work on Fluid Structure Interaction (FSI) of PPOOLEX experiments was continued by analysing high-speed videos and pressure measurements on rapid
condensation of vapour bubbles. The effect of finite condensation rate on the bubble collapse and the scaling of the laboratory experiments to BWR plant were studied. FSI calculations of the BWR plant were performed with the acoustic model of ABAQUS. The stochastic analysis of pressure loads from multiple vent pipes was continued.

**Specific goals in 2012**

The chugging phase of blowing vapour into a pressure suppression pool has been studied. In the chugging phase, rapid periodic condensation of vapour bubbles occur at the outlets of vent pipes. The CFD model for chugging was further developed. In CFD modelling, the goal has been to obtain correct collapse speeds of vapour bubbles and correct penetration of liquid water in the vent pipes. The new possibilities of obtaining high speed videos in the vent pipe in the PPOOLEX experiments has been utilized in comparisons. Simulations of the chugging phase of a PPOOLEX experiment were performed with a short time step in order to achieve improved resolution in the simulations.

The PPOOLEX experiment MIX-03 was calculated by using the new version of the direct-contact condensation model. Some improvements in the qualitative and quantitative behaviour of the condensation rate were achieved. Stronger condensation rates were obtained, which is in better agreement with the experiments than earlier simulations. The calculated behaviour of the formation and condensation of steam bubbles is illustrated in Figure 2.4.4.9.

Modelling of the rapid bubble collapse has been refined by taking into account the effect of the finite condensation rate. This was done by fitting the calculated collapse time and pressure load with the experiments. The simulations were compared to the high-speed video images and pressure measurements from the experiments. The scaling of the pressure load amplitude and duration from the laboratory experiment to the BWR was also studied by dimensional analyses. Scaling properties obtained from the dimensional analyses were compared with earlier numerical results.

Two-way coupled FSI calculations have been performed for a model of a sector of BWR containment. The loads in a BWR were calculated with an acoustic model of full containment, i.e., 16 vent pipes. Stochastic analysis of the loads originating from multiple vent pipes was used to analyse the loads in the chugging phase. The pressure source statistics, such as load amplitude and delay between the vent pipes, was determined from experiments and data available in literature. Simulation of the loads and structural response in the BWR geometry was performed.

Simplified model geometry of containment was created to enable the studying of the structural response during multiple chugging events with varying desynchronization time. The computations were performed with and without material damping. It was observed that desynchronization time in the loading decreases the maximum stresses significantly for the case where material damping was taken into account. In turn, the difference between the maximum stresses was very small between synchronized and desynchronized loading in cases where material damping was not considered. This resulted from the fact that the containment was still in motion caused by the previous group of chugs when next group of chugs took place. The von Mises stress as a function of time in the model in which material damping is taken into account is presented in Figure 2.4.4.10 for synchronized and desynchronized loading.
Figure 2.4.4.9. Calculation of the formation and condensation of a steam bubbles at the vent pipe outlet in the PPOOLEX experiment MIX-03.

Figure 2.4.4.10. Von Mises stresses at elements in which the maximums occur for a) synchronized loading with material damping, b) desynchronized loading with material damping.

**Deliverables in 2012**

- CFD simulation of chugging in a PPOOLEX experiment
- Calculation of collapsing vapour bubble in a PPOOLEX experiment and scaling analyses of the pressure loads
- FSI calculation of chugging with a BWR model with multiple vent pipes
- Report on the CFD and FEM calculations.

**2.4.5 Improvement of PACTEL facility simulation environment (PACSIM)**

The main objective of the PACSIM project has been to improve the simulation environment of the PACTEL facility with the TRACE thermal hydraulic code. The Finnish Radiation and Nuclear Safety Authority, STUK, has required an independent tool to support safety and licensing analysis and decided to use the TRACE code. The use of the TRACE code enhances
the preparedness to give analysis support and improves education in computational thermal hydraulics. This project aimed to develop a complete TRACE code model of the PACTEL facility and use the model for validation calculations at Lappeenranta University of Technology. PACSIM project started at 2008 and was completed at 2012 and hence has been operating during SAFIR2010 and SAFIR 2014 research programmes. The TRACE code has been used for preparation of PACTEL facility model with horizontal steam generator and also for preparation of PWR PACTEL model with vertical steam generator. Various validation cases have been modeled and calculated.

The validation cases consisted among others of small break LOCA experiments with different break sizes as well as cases of primary-to-secondary side leakage and also one ATWS case. The project also summarized the outcome of validation knowledge received from different calculations. This information can be utilized for modeling of the full-scale VVER-440 model preparation, which has and will be carried out outside the SAFIR programmes. The results of PACSIM project has been and will be also utilized in PAX project, where PWR PACTEL model has been used and developed further.

**Specific goals in 2012**

Specific goals in 2012 were validation further the full TRACE-model, which has been prepared in PACSIM project during 2008 and developed further during 2009-2011. During 2012 the project was continued concentrating on the modeling and calculations of ATWS-experiment. The main interest will was focused on ATWS-32 experiment. This experiment simulated the control rod withdrawal from the core, when it was operating at very low power range, almost zero power. Hence, prompt power increase occurred and pressurizer safety valves had to be opened to prevent the over pressurization of the primary side. The ATWS-32 was then modelled with the TRACE code using core power as boundary condition. The three relief valves were included in the calculation model. The main part of the used modeling setup for the ATWS-32 experiment is presented in Figure 2.4.5.1. The resulted pressure propagation in TRACE calculation compared to the experiment is presented in Figure 2.4.5.2.

Another goal in the PACSIM the project was to conclude the validation knowledge throughout the project. This subtask aimed to gather the knowledge and experience of the TRACE code model studies, which has been gained in the PACSIM project since 2008. Table 2.4.5.1 presents the list of calculated PACTEL experiments in the frame of PACSIM project.
• Figure 2.4.5.1. TRACE/SNAP modeling of PACTEL facility for calculations of ATWS-32 experiment. Loop one visible.

Figure 2.4.5.2. Primary and secondary side pressures in experiment ATWS-32 vs. TRACE calculation.
Table 2.4.5.2. PACTEL experiments calculated with TRACE code in the frame of PACSIM project.

<table>
<thead>
<tr>
<th>Experiment name</th>
<th>Objectives</th>
</tr>
</thead>
<tbody>
<tr>
<td>FLT-04, -08 and -11</td>
<td>Definition of pressure losses in normal and reversed directions</td>
</tr>
<tr>
<td>HL-22, HL-23</td>
<td>Definition of heat losses for whole facility and pressurizer</td>
</tr>
<tr>
<td>LOF-10</td>
<td>Calculate LOF-experiment for both stand-alone and full models</td>
</tr>
<tr>
<td>SIR-21, SIR-23</td>
<td>Stepwise inventory reduction investigating natural circulation modes in full and low pressure ranges</td>
</tr>
<tr>
<td>SBL-30</td>
<td>Small break LOCA 1 mm break, pressurizer isolated in transient</td>
</tr>
<tr>
<td>SBL-31</td>
<td>SB LOCA 2.5 mm break, ACCU performance and secondary side feed and bleed</td>
</tr>
<tr>
<td>SBL-31</td>
<td>SB LOCA 3.5 mm break, ACCUs, HPIS and secondary feed and bleed</td>
</tr>
<tr>
<td>IMP-06</td>
<td>7.8 mm break, lowered ACCU set point pressure and initiation of LPIS</td>
</tr>
<tr>
<td>PSL-10, PSL-11</td>
<td>PRISE experiments, collector ruptures of 5.5 mm and 2.5 mm</td>
</tr>
<tr>
<td>ATWS-32</td>
<td>Simulation of control rod withdrawal from the core, power feedback</td>
</tr>
</tbody>
</table>

Deliverables in 2012

- PACTEL experiment ATWS-32 was calculated with the TRACE model. This experiment described the simulated the control rod withdrawal from the core, when it was operating at very low power range, almost zero power. Hence, prompt power increase occurred and pressurizer safety valves had to be opened to prevent the over pressurization of the primary side. The overall behavior could be modeled satisfactorily until the moment, when flow stagnation took place in the experiment. This event was not found in the TRACE calculation. This discrepancy caused that the rest of the calculation results were not fully comparable with the experiment. Despite of the anomalies the calculated overall natural circulation flow rate appeared to agree rather well with the experiment value. The validation report described the results in more detail.

- The validation knowledge was gathered from the TRACE code modeling. The TRACE code has been used for preparation of PACTEL facility model with horizontal steam generator with necessary auxiliary systems and control devices. The TRACE code was also used for preparation of PWR PACTEL model with vertical steam generator. For PACTEL facility model 15 experiments were calculated to find out pressure and heat losses and to validate the model against various transients. The validation cases consisted among others of small break LOCA experiments with different break sizes as well as cases of primary-to-secondary side leakage and also one ATWS case. The validation calculations succeeded rather well to model large variety of transients. The steady state periods were modeled accurately almost without exception. Two phase break flows appeared to be the most challenging phenomena for the TRACE code to model. Some discrepancies took place in modeling of flow stagnation during natural circulation. The separate report concluded the knowledge of the validation results in more detail.
2.4.6 PWR PACTEL experiments (PAX)

The objective of the project is to utilize the new PWR PACTEL test facility in an effective way in nuclear safety research in Finland and internationally. The first step was the launching of the PWR PACTEL calculation benchmark in the PAOLA project, funded by Tekes, TVO and Fortum. The PWR PACTEL test program continued in 2012 with one U-leg draining experiment series and by participating in the OECD/NEA PKL Phase 3 project.

Specific goals in 2012

Specific goals in 2012 included PWR PACTEL U-leg draining experiment and participating in the OECD/NEA PKL Phase 3 project. The U-leg draining phenomenon is well known and it has been observed on different test facilities. Analyses of EPR type nuclear power plant have shown that the peak cladding temperature depends not only on the core uncover depth and duration but also on the number of U-legs drained. The U-leg draining phenomenon is mainly driven by the size and location of the break.

The experiments were proposed by TVO to validate the EPR FSAR analyses. A series of experiments and related computer analyses were performed to determine the effect of break size on U-leg draining during intermediate break loss-of-coolant-accident conditions and to test the suitability of the PWR PACTEL facility for the U-leg draining studies. Both loops and the pressurizer were used in the experiments. Two emergency core cooling systems were also used. An orifice plate was used to simulate the break size and to control the loss rate. Parameter values, boundary conditions as well as initial break sizes for the experiments were chosen based on the results of the pre-test calculations with the TRACE code.

In the experiments both U-legs opened simultaneously. As a result, the main driving force leading to U-leg clearing, namely the pressure difference across the U-leg, was strongly reduced and the second U-leg did not remain open (Figure 2.4.6.1 and Figure 2.4.6.2). The U-leg that remained open was not the same in the LSC-02 and LSC-03 experiments. In these experiments the effect of break size on U-leg draining could not be verified as the safety limit for heater rod temperature was exceeded in one experiment and it had to be terminated too early. Despite of this these experiments showed that the PWR PACTEL facility is suitable for U-leg draining studies with some limitations characteristic to the facility. The experiments also showed that small breaks can be more serious for the core than bigger ones and confirmed the results of EPR analyses.
The simulations of the LSC-03 experiment were performed with the APROS and TRACE codes. The U-leg draining was visible in both simulations.

Two PWR PACTEL experiments have been proposed to be included in the OECD/NEA PKL Phase 3 project; dry-out of the SG secondary side at high core power (ATWS) and cool down procedures in presence of isolated steam generators (comparison with PKL tests in two loop configuration). The experiment on cool down procedures in presence of isolated steam generators was planned in 2012. One preliminary test was carried out to test the systems and experiment procedure. The APROS and TRACE codes were used in the pre-test calculations to help the experiment planning.
Deliverables in 2012

- A series of experiments proposed by TVO and related computer analyses were performed to determine the effect of break size on U-leg draining during intermediate break loss-of-coolant-accident conditions and to test the suitability of the PWR PACTEL facility for the U-leg draining studies.

- The first PWR PACTEL experiment in the OECD/NEA PKL Phase 3 project was planned and one preliminary test was carried out to test the systems and experiment procedure.

- The PWR PACTEL benchmark article was offered to be published in Annals of Nuclear Energy.

2.4.7 Modelling of pressure transients in steam generators (SGEN)

Three-dimensional CFD simulation tool is developed for the modelling of large pressure transients on the secondary side of steam generators of nuclear power plants. The models for evaporation and condensation are improved for simulations of rapid pressure transients. The model is applied to a transient in a horizontal steam generator of a VVER-440 nuclear power plant and to the vertical steam generator of the PWR PACTEL facility. In future, the developed two-phase models can also be applied to other situations, where rapid evaporation or condensation occurs.

Specific goals in 2012

Three-dimensional CFD models for the secondary side of horizontal and vertical steam generators have previously been developed. The tubes of the primary circuit are modelled with the Apros system code and the outer wall temperature is interpolated into the ANSYS Fluent model of the secondary side. In the Fluent model, the geometry of the primary circuit is not described in detail but the drag force and the heat transfer are described with a porous media model.

In 2011, the model for the secondary side of steam generator was partly rewritten. The evaporation and condensation models have been modified to better describe also transient situations. An interface to Apros steam tables has been included in the model. The user-defined real gas model of Fluent was used for describing the properties of the compressible vapour.

When large pressure changes occur on the secondary side of the steam generator, heat transfer from wall structures is important. The submodels developed for wall condensation and evaporation were adapted into the steam generator model. Robustness of the solver was tested and improved.

The implemented models were tested by calculating a stationary state natural circulation experiment NC-10 performed with the PWR PACTEL facility. The circulation of water on the secondary side of the steam generator was analysed in detail. The main circulation in the experiment was from the hot riser to the steam dome, where the liquid-water returned via hot downcomer to the hot riser. The flow through narrow gaps between the hot and cold downcomer was analysed in detail.

The PWR PACTEL experiments RF-02 and RF-04 were recalculated, where pressure drop and pressure rise occurs on the secondary side. In the experiment RF-02, the pressure on the
secondary side was reduced from 40 bars to 30 bars. The pressure drop lead to rapid steam
generation on the secondary side. The increase of void fraction is illustrated in Figure 2.4.7.1.
In the experiment RF-04, the pressure on the secondary side increased from 20 bars to 25
bars. The pressure increase leads to reduction of void fraction on the secondary side and
collapse of the water level.

A loss-of-feed-water transient in the horizontal steam generator of the Loviisa NPP was
calculated. The geometry of the three-dimensional model for the steam generator was updated
by including a model of steam dryer in the CFD calculation. The chosen transient was first
calculated by using an Apros power plant model. The results of the analysis were used as
boundary conditions for detailed Apros model of the steam generator, where the temperatures
of the primary tubes were solved. The calculated temperatures of the primary tubes were used
as a boundary condition for three-dimensional CFD calculation of the secondary side of the
steam generator. In addition, pressure and mass-flow boundary conditions for the CFD model
were achieved from the APROS power plant model.

Figure 2.4.7.1. Calculated increase of void fraction on the secondary side of the PWR
PACTEL steam generator, when pressure decreases from 40 bars to 30 bars.
Figure 2.4.7.2. Calculated change in void fraction on the secondary side of the VVER-440 steam generator, when pressure increases from 45.3 (100 seconds) bars to 52 bars (150 seconds) and decreases to 47.5 bars (200 seconds).

In the simulated transient, feed water pumps trip after 100 seconds. This leads to rapid changes in the secondary side pressure and steam generation. As the feed water is lost, the water mass in the steam generator decreases. In Figure 2.4.7.2, void fraction is presented in the middle of the steam generator.

**Deliverables in 2012**

- Report on validation calculation of the rapid phase change models.
- Report on the calculation of a PWR PACTEL experiment.

**2.4.8 Uncertainty Evaluation for Best Estimate Analyses (UBEA)**

The objective of the project is to study the use of Best Estimate Plus Uncertainty (BEPU) approach to thermal hydraulic safety analyses of nuclear power installations, with focus on the evaluation of the input parameter uncertainties needed in the BEPU analyses. The BEPU approach is an alternative to the conservative approach that has been conventionally used for the licensing analyses of nuclear power plants, and it should result in more realistic predictions of the plant behaviour in various transient and accident scenarios.

BEPU analyses require that the uncertainties of all the relevant input parameters are quantified by associating them with a probability density function with known parameters. For some parameters the uncertainties can be directly measured, while for some other parameters the uncertainties can only be estimated through comparing simulation results to experimental data. Methods that can be used for this latter-kind of quantification are studied in the OECD/NEA/WGAMA PREMIUM benchmark project.

**Specific goals in 2012**

Specific goals in 2012 include a revised Best Estimate Plus Uncertainty analysis of the ISP-50 Problem, participation to the PREMIUM benchmark project and assessing applicability of the APROS code in Deterministic-Probabilistic Safety Assessments (DPSA).
The OECD International Standard Problem number 50 (ISP-50), which had previously been calculated at VTT with the APROS code, was reanalysed using the BEPU approach. The list of varied parameters was created based on the recommendation of the BEMUSE project. The BEPU analysis of 93 runs was applied three times to the transient scenario to provide for internal comparison of the method and the achieved results. The output parameter variations in the BEPU analyses were all in accordance with one another and no dramatic differences were found in this respect. Sensitivity studies were performed in the form of limit runs where each input parameter was varied on the ranges of their PDF's. Also, statistical correlation coefficients were applied to the results of the BEPU analyses for input parameter sensitivity analysis purposes. Contrary to the similarity of the BEPU analyses results the sensitivity analyses provided conflicting results to some degree, but this was to be expected with the ratio of the amount of input parameters to simulation runs being so high (26/93). Because of the unreliability of the sensitivity studies only parameters with very high influence were revealed, even with a ranking table that integrated all the different sensitivity analyses.

The ISP-50 analysis provided a basis for future research into best-estimate methods using APROS and laid out the general workflow for a BEPU analysis. What was found out was that to achieve reliable results with the BEPU method it is extremely important to use a model that is able to reproduce the real situation very precisely. A recommendation was made of applying the BEPU method to APROS validation cases to provide more information on the usage of the method with APROS cases.

The PREMIUM (Post BEMUSE Reflood Models Input Uncertainty Methods) benchmark exercise is organized between the years 2012 and 2014. In the benchmark, different methods for quantification of the uncertainties of the physical model parameters are put into use in calculation of reflooding tests from two separate test facilities: FEBA and PERICLES-2D. VTT Technical Research Centre of Finland participates in the PREMIUM benchmark using the APROS code. During the year 2012, the focus in the benchmark project was to identify all the influential parameters that affect the outcome of a reflooding simulation of the FEBA facility.

An initial list of all parameters that could be influential to the main responses of interests such as cladding temperatures, rewetting time instant, quench front elevation and pressure distribution within the channel was first drafted. The initial list included practically all physical parameters of the experiment that have been modelled in the APROS code. Next, the overall influence of all parameters was evaluated by simulating the reflooding experiment numerous times, with varying a single uncertain parameter at a time, at the limits of its predetermined PDF. To determine which of the parameters on the initial list were influential, a set of criteria related to the maximum cladding temperature and the rewet time variation, proposed in the benchmark specification, was applied. In the end 30 out of the 40 parameters on the initial list were pruned out.

With the initial quantification of the influential input parameters, the experimental results were not enveloped by the bounding curves obtained as a result of the BEPU simulation, and it seems that APROS underestimates the rise of the cladding temperatures, and the quench times. Examination on the probable cause for the observed behaviour suggested that the interfacial heat transfer and friction models used above the quench front are in their current form inadequate to properly describe the reflooding phenomena at least in the case of the FEBA experiment.
A small literary review was carried out to gain insight on how system codes have been applied to Deterministic-Probabilistic Safety Assessment, and to see if there would be benefit in using APROS and the accompanying TestingStation tool in dynamic PSA analyses or in DPSA in general. A short review report on possible application of APROS and Testing Station to Deterministic–Probabilistic Safety Assessment was written, and judging by the results of the review work, it would be possible to develop a dynamic safety analysis tool around APROS and Testing Station software, but the capabilities of the APROS code would limit the scope of application to Level 1 PSA. This would not require any PSA-specific code changes in the APROS code, but on the other hand the development of a PSA module to the Testing Station tool would require considerable amount of work. Added to the fact that the resulting tool would only be applicable to Level 1 PSA, this development may not be worthwhile.

![Graphs showing some results from the three BEPU analyses of the ISP-50: the downcomer liquid level (top) and the cladding temperature (bottom). Portrayed are the bounding curves, the best estimate curve, the conservative curve, the average and median curves, along with the experimental results and previous ISP-50 analysis results for comparison.](image)

**Figure 2.4.8.1.** Some results from the three BEPU analyses of the ISP-50: the downcomer liquid level (top) and the cladding temperature (bottom). Portrayed are the bounding curves, the best estimate curve, the conservative curve, the average and median curves, along with the experimental results and previous ISP-50 analysis results for comparison.

**Deliverables in 2012**

- The ISP-50 problem was analyzed using the Best Estimate Plus Uncertainty Approach
- The influential input parameters of the PREMIUM benchmark were identified through calculation of FEBA reflooding experiment
- Applicability of APROS in Deterministic-Probabilistic Safety Assessments was studied
2.5 Severe accidents

In 2012 the research area "Severe Accidents" consisted of five projects: Core debris coolability (COOLOCE), Chemistry of fission products (FISKE), Thermal hydraulics of severe accidents (TERMOSAN), Transport and chemistry of fission products (TRAFI) and Reactor vessel failures, vapour explosions and spent fuel pool accidents (VESPA).

2.5.1 Core debris coolability and environmental consequence analysis (COOLOCE-E)

The COOLOCE-E project focuses on two research topics in the area of severe accident management, core debris coolability and the assessment of the environmental consequences of a severe accident. The main objective of the project is to conduct debris bed coolability experiments important in the safety assessment of the Finnish boiling water reactors. The experimental data are used for simulation code validation and development and reducing the uncertainties related to debris coolability in power plant scenarios. Collaborative efforts with VTT and KTH are performed aiming for comprehensive experimental and analytical investigations of the key issues of ex-vessel debris bed coolability and bed formation. The goal of the environmental consequence assessment is to estimate the radiation doses caused by the Fukushima accident. The calculations will be used for code verification and validation and to introduce a new researcher to the topic.

Specific goals in 2012

Specific goals of the debris coolability research in 2012 were 1) to conduct experiments with irregular gravel to verify earlier results and to investigate the possible differences specific to simulant materials and test facilities, 2) to conduct an experiment to scope the effect of pool subcooling on dryout power, c) to measure the dryout power in a mound-like debris bed (water infiltration through top and lateral surfaces) and d) to continue the coolability simulations with 2D and 3D two-phase flow solvers for code validation and development purposes. The modelling work is an integral part of the project because reactor-scale assessment is done by severe accident simulation codes.

The objective of the environmental consequence subproject was to perform atmospheric dispersion simulations of the radionuclide release of the Fukushima Dai-ichi accident, and compare the results obtained by different codes to the monitored values of deposition and dose rates in the vicinity of the plant site. Training of a new expert in this highly topical area with the help of senior researchers was another objective.
Figure 2.5.1.1. Test beds of the experiments in 2012: cylindrical bed with irregular gravel for a top flooding experiment (left) and with beads for a “mound-like” bed experiment (right). Note that in the photo on left, the cylindrical sidewall used in the experiment is not yet attached.

Figure 2.5.1.2. VALMA simulation of Cs-137 deposition in Japan on March 20, 2011.

Deliverables in 2012

- The COOLOCE-8 experiment was conducted with irregular alumina gravel. The results clarified the uncertainties related to the simulant material and to the measurements by test facilities with different types of heating arrangements and test vessels. It was found that the dryout heat flux measured for the alumina gravel was low compared to the experiments with spherical beads and slightly lower compared to the earlier experiments with the STYX facility.

- The COOLOCE-9 experiment was performed with initially subcooled pool (no pre-test heat-up sequence). The results are rather preliminary but they strongly suggest that if the
pool into which the debris bed is submerged is not completely saturated, coolability is increased.

- The COOLOCE-10 experiment investigating the mound-like bed geometry was performed. It was found that the dryout power and coolability is significantly increased as a result of the lateral flooding. Compared to a cylinder with top flooding only, the increase is more than 50%.

- Test facility modifications necessary for the new experiments were made, including the installation of the cylindrical sidewall supported with a wire net, replacing damaged heaters and some of the old steam line valves and the calibration of a new power meter.

- A feasibility study scoping the possibilities to install a void probe to measure void fraction and fluid velocity in the test vessel was performed. This data would complement the thermocouple arrangement used to detect dryout, and could be utilized in code validation. However, it was found that the acquisition of such a system is not currently feasible because its use and data analysis would be require financial and personnel resources not available at the moment.

- MEWA simulations that investigated the sensitivity of dryout power to physical model parameters, particle diameter and porosity, in the cylindrical bed experiments (COOLOCE-3 – 5 and COOLOCE-8) were run. The PORFLO modeling of the debris beds with conical and cylindrical geometries were continued using the new version of the code. The models programmed for PORFLO were included as user-defined functions in the widely-used commercial CFD solver FLUENT. The goal of this was to verify the implementation of the physical models in the different 2D and 3D codes and to otherwise test the performance of the codes to e.g. sort out numerical problems.

- Separate measurements to determine effective particle diameters of the debris simulant materials used at VTT were performed at KTH (Kungliga Tekniska Högskolan, Sweden) laboratories. Planning of the experimental and analytical activities was done jointly with KTH and information was exchanged within the NKS/DECOSE project.

- A literature search was done to collect up-to-date material of the Fukushima accident and the radionuclide release caused by it. The material consisted of measured air concentrations, dose rates and depositions and published simulation data. The weather data during March 10 - May 9, 2011 was obtained from FMI (Finnish Meteorological Institute), based on the weather prediction model of European Centre on Medium-Range Weather Forecast (ECMWF). VALMA simulations were done with a preliminary comparison to the data reported in the literature.

- One journal article and one conference paper addressing the work on debris bed coolability were published. Three research reports were written describing the COOLOCE experiments and simulations and one report describing the Fukushima dispersion data and simulations.

### 2.5.2 Chemistry of fission products (FISKE)

Results of various experiments have shown that sump solution pH has major effect in fission products release rate. Nitric acid is a principal radiolytic compound produced in large, and its production is another important problem concerning pH of solutions, owing to its chemical properties of being a strong acid and a strong oxidizing agent. VTT participate OECD/BIP2
project. The information from that project is useful to compare our own experimental iodine test results, especially the modelling part could be helpful to understand the phenomena.

From the beginning of core melting up to the control of the accident by stopping and cooling the corium, the knowledge of corium physical properties versus temperature is essential to predict possible scenarios with a view to managing the accident. Viscosity plays a major role in many phenomena such as core melt down, discharge from reactor pressure vessel, interaction with structural materials and spreading in a core catcher. In addition, interactions between corium low-volatility fission products (etc. Sr, Ba, La and Ru) and concrete are important to understand.

**Specific goals in 2012**

A new pool chemistry simulation model, ChemPool, is ready to testing and use. It is a standalone program using ChemApp thermodynamic library directly for equilibrium calculation (which ChemSheet also uses internally). ChemPool program has been integrated more closely to MELCOR by making it read and interpret MELCOR input files. ChemPool will then automatically calculate the Chemical conditions like pH values in the pools by using the saved results from the MELCOR simulation. It has been tested with BWR accident case and now it will be tested PWR accident case.

Earlier advanced containment experiments (ACE) project shows that there might be the fission product release during corium and concrete interactions. The comparison between CHEMSHEET and GEMINI2 calculations will be done in year 2012 using ACE project experimental data.

In the third goal is to calculate the 2D temperature profile for the Core Catcher with different decay heat values and then to calculate mass fraction of liquid in corium in concrete mixture at the solved temperature.

**Deliverables in 2012**

ChemPool model has been updated to allow mixing of flow components between the pools (both water pool and atmosphere), this includes acids and bases and aqueous phases and gas phase (mainly vaporized HCl and HI). The investigated accident scenario was a total blackout and a big break LOCA in the cold leg of a PWR. This accident results in core melt and a significant fraction of the radioactive material is released to the containment. Input data for radiation dosage calculations was obtained by the MELCOR calculation, in which spreading of the radioactive material in different volumes of the containment was determined. Time dependent isotope concentrations were obtained by the ORIGEN2 code. The nitric acid formation were calculated and pHs of water pools were calculated using a new developed ChemPool program. In these results only the formation of nitric acid in water phases were calculated.

ChemSheet and GEMINI2- calculations for the L8 ACE- test case show that the amounts of gaseous Sr, Ba and La are very small or non-existent; the amounts of species actually observed in aerosol samples were in same level. The results from GEMINI2 and ChemSheet are practically identical except for Fe, Cr and Ni that have small differences. FactSage has a limitation considering the number of interaction parameters for a solution phase. Because of large amounts of metal elements in the thermodynamic system, not all of them could be present in the liquid phase at the same time which is disadvantage.
Steady state heat conduction model requires that no transient changes take place in the core catcher. This means that all concrete must have been melted (except at the edges where the temperatures are lower than the solidus temperature of the concrete) and mixed with the corium. Further in the model it is assumed that the metallic and the oxidic phases are separated from each other due to their different densities, and that there is no mixing taking place in the core catcher where it is in liquid state. All these assumptions are simplifications and do not hold in the real process. Model showed that the maximum temperature at 2 hours in the core catcher was 1984 K, and at 6 hours 1564 K. Model calculated situation at 2 hours with both the metallic and oxidic liquid phases on top. If the oxidic phase was on top then the maximum temperature in the core catcher was 1864 K.

![Figure 2.5.2.1. 28w-% Zr oxidized: Temperatures at 2 hours (decay heat 44.7 MW).](image)

**2.5.3 Thermal hydraulics of severe accidents (TERMOSAN)**

The objective of the project was to improve modeling capabilities on severe accident thermal hydraulics. MELCOR is Finland’s main severe accident analysis tool. MELCOR license for Finnish nuclear energy organizations is obtained from the U.S.NRC with the CSARP agreement in the frame of the TERMOSAN project. MELCOR was used for developing a model of the Fukushima accident. The integral code ASTEC has been adopted in the frame of the project to augment VTT's capabilities to provide a larger variety of independent tools for severe accident studies. Two international OECD NEA research programs, THAI-2 and SERENA-2, were participated in the frame of this project.
**Specific goals in 2012**

MELCOR is an integral severe accident analysis code, developed in the USA. In the frame of the TERMOSAN project, Finland participates in CSARP (Co-operative Severe Accident Research Program). This gives us a license for latest versions of MELCOR and the right to participate in the annual CSARP meeting.

The first version of VTT’s MELCOR model of Fukushima Daiichi unit 1 accident was developed. The new MELCOR version 2.1 was used. This was the first full-plant model developed with the new code version at VTT. Until now the previous 1.8.6 version had been used in all VTT’s full-plant calculations, but some experiments had been modeled with the new code version. Available plant data about the Fukushima plant is scarce at the moment. Information was collected from various Japanese publications. Missing pieces of plant data were taken mainly from the Peach Bottom plant, which is similar to the Fukushima plant but larger. It is intended to update the model when more plant data becomes available.

The accident started with the earthquake at 14:46 Japanese time on March 11, 2011. This was set as time zero in the calculations. The earthquake triggered a reactor scram. The reactor was cooled with the isolation condenser for 51 minutes, until the tsunami inundated the site, submerging the emergency diesel generators and much of the emergency power distribution systems, causing a total loss of AC and DC power. In order to simplify this first version of the MELCOR model, the calculation starts at the time of tsunami arrival. This eliminates the need to simulate the isolation condenser. According to the calculation results, the core uncovery starts 2 h 35 min after the earthquake, and the fission product release from the core begins at 3 h 59 min. The core degradation begins at 5 h 15 min, when the control rods start to collapse. The first fuel rod collapse occurs at 5 h 37 min, and eight minutes later there is a major fuel relocation to the lower plenum. At 6 h 19 min all water has evaporated from the RPV, and the RPV lower head failure is calculated to occur at 9 h 35 min.

State of the Fukushima reactor at four instants of time is illustrated in figure 2.5.3.1. The fuel rods in the innermost ring collapse first because the decay heat power is highest in the center of the core. Figure 2.5.3.2 shows the calculated containment pressure, compared with measurements. The pressure increases rapidly when the RPV fails at high pressure at 9.6 h, and the match with the two measurement points at that time is good. At 12 h the pressure is underestimated by 0.17 MPa. After 15 h the calculated pressure matches very well with the measurement because the containment leak model and the venting flow path were tuned for that purpose.

After the RPV failure molten corium is discharged to the containment and the concrete basemat begins to melt. The calculation gives about 1.2 m concrete ablation at 50 h, and the ablation still continues. Uncertainties at this phase of the accident are so large that this value is little more than a guess. The calculation gives 19 % hydrogen concentration in the top floor of the reactor building at the time of the explosion. This is clearly an explosive concentration, but again the uncertainties of the calculation are very large. Also the radioactive releases to the environment were calculated. It is estimated that, during 50 h, the release fraction was 4E−1 of core inventory for noble gases and 3E−4 of core inventory for cesium. Uncertainties in these numbers are larger than factor of ten.
Lappeenranta University of Technology is starting PCCS experiments on passive containment cooling systems with horizontal tubes. In the frame of the TERMOSAN project, an ASTEC input deck of the PPOOLEX PCCS test facility was created. This model will be used in 2013 to simulate the experiments.

In the frame of the TERMOSAN project, Finland participated in two OECD NEA research projects. The SERENA-2 project investigated steam explosions. The THAI-2 project investigates aerosol and iodine issues and hydrogen behaviour. In the TERMOSAN project, THAI HD-30 experiment on hydrogen combustion during spray operation was calculated.
with Fluent. Various turbulence and burning velocity models and laminar flame speed settings were studied. Cases with and without spray were reproduced reasonably, but with different models. None of the models gives reasonable results for both cases.

**Deliverables in 2012**

- The MELCOR annual license fee was paid, giving the license to use MELCOR for all Finnish nuclear energy organizations.
- The CSARP/MCAP meeting was attended and a presentation on MELCOR modeling of passive containment cooling system experiment PANDA T1.1 was given. A travel report was written.
- A MELCOR model of Fukushima unit 1 accident was developed.
- An ASTEC input deck for the PPOOLEX PCCS test facility was created.
- OECD SERENA-2 program meeting and final seminar were attended and a travel report was written.
- Two OECD THAI-2 program meetings were attended and travel reports were written.
- THAI HD-30 experiment on hydrogen combustion during spray operation was calculated with Fluent.

### 2.5.4 Transport and chemistry of fission products (TRAFI)

The objective of the project is to study the behaviour of fission products in a severe accident conditions. In particular, the aim is to increase understanding of revaporisation and transport of iodine in primary circuit and containment of a nuclear power plant. The primary circuit study has been conducted in close co-operation with IRSN Cadarache research centre for the determination of iodine chemistry. The objective of the primary circuit study at VTT is to determine iodine compounds released due to the reactions on the surface of primary circuit piping. At the same time IRSN focus in the gas phase chemistry of iodine in similar experimental conditions. The measurements with EXSI-PC provide information on high temperature chemistry and facilitate validation of for example iodine chemistry codes. Radiolytical oxidation of gaseous iodine by various radiation sources in containment conditions is studied using EXSI-CONT facility. The objective is to verify the possible formation of iodine oxide aerosol particles. Another aim is to identify the reaction product species. The desorption of gaseous iodine from iodine containing aerosol deposited on different containment surfaces is studied in co-operation with FISKE project and Chalmers University of Technology. The deposition of aerosol particles in a differentially heated cavity in turbulent natural circulation flows is studied with DIANA facility together with Paul Scherrer Institut. Measurements conducted at VTT are applied in validation of DNS and LES simulations carried out at PSI.

International collaboration is also conducted by participation in the work of Phebus FP, International Source Term Programme (ISTP) and OECD/NEA STEM programs.
Specific goals in 2012

The main goals in 2012 were to study the behaviour of iodine in primary circuit and containment conditions. In primary circuit experiments the effect of reactions on primary circuit surfaces on the release and transport of iodine was studied. The primary circuit experiments were conducted using updated EXSI-PC facility. The modifications done to the facility during 2011 have significantly improved the monitoring and data logging capabilities of the set-up. With the updated facility, comparison of the measurement data acquired from the different measurement devices could be done more accurately.

Another goal was to study the radiolytical oxidation of gaseous iodine when exposed to beta radiation in containment conditions. The aim was to build a test facility and to verify the formation of aerosol particles. Another task was to identify the reaction products.

As a third goal was to study the behaviour of iodine, deposited as iodine oxide or caesium iodide aerosol, on various containment surfaces. The objective was to determine the influence of gamma radiation dose, humidity and temperature on the desorption of iodine from iodine containing aerosol deposition on fresh or aged paint, stainless steel, copper, aluminium and zinc surfaces. The particles deposited on surfaces were analysed with several techniques. The transport of released radiolabelled iodine (I$^{131}$) between gaseous phase and liquid phase was followed as well. In some experiments, the samples were placed directly in a liquid phase.

As a last goal, the particle depletion in a differentially heated cavity with turbulent natural convective flows was experimentally studied with DIANA facility.

Deliverables in 2012

- In the primary circuit studies the source of iodine was CsI or AgI powder which was evaporated at 400°C, 550°C and 650°C on ceramic or oxidized stainless steel surface under Ar/H$_2$/H$_2$O atmosphere. The surface of the reaction furnace tube, made of stainless steel, was pre-oxidized before the experiments. These experiments showed that, when CsI was used as a precursor with Ag, B$_2$O$_3$, MoO$_3$ or Ag+MoO$_3$, a significant amount of the released iodine was in gaseous form. This indicates that the reactions at primary circuit surfaces contribute to gaseous iodine transport in severe nuclear accidents. Especially, it was observed that in relatively low temperature, the released iodine was mostly in gaseous form, if some of the materials mentioned above were present (Figure 2.5.4.1). As CsI and B$_2$O$_3$ reacted together, caesium borate glass was found to be formed on the reaction crucible surface.

- Studies on the radiolytical oxidation of gaseous methyl iodide by beta radiation verified the formation of aerosol particles (Figure 2.5.4.2). Their diameter ranged from ~10 to 50 nm and a fraction of particles with a diameter higher than 100 nm were also observed in some experiments. Particles were highly water soluble and volatile; therefore the analysis of particles with SEM-EDX was difficult. The formation of CO$_2$ seemed to be enhanced when the duration of radiation exposure was increased.
Figure 2.5.4.1. The transported iodine, caesium and silver mass concentrations as gaseous and aerosol form in primary circuit experiments. Iodine transported mainly as aerosol particles when CsI + Ag precursor mixture was heated to 650 °C (experiment A). As temperature was decreased to 400 °C (experiment B), all the released iodine was in gaseous form. Condition A has Ar/H₂O atmosphere whereas conditions B and C contain also H₂.

Figure 2.5.4.2. A SEM micrograph of solid, partially dissolved and dissolved iodine oxide particles collected directly from a gas phase on a nickel/carbon grid. Radiolytical oxidation of gaseous methyl iodide by beta radiation in oxygen atmosphere led to a formation of particles with a diameter of ~10 to 50 nm. The formed particles were highly water soluble and volatile.
• The analysis of iodine oxide and caesium iodide particles deposited on various containment surfaces was conducted with e.g. SEM + EDX, XPS and RAMAN methods. Iodine oxide particles produced at 50 °C, 100 °C and 150 °C reacted with the humidity of air and decomposed to iodic acid and elemental iodine. Iodine from IOx and CsI deposits reacted with the studied surfaces, e.g. the formation of water soluble and insoluble metal iodides was observed. Humidity seemed to have a major effect on the increase of gaseous iodine desorption from the deposits.

• The temperature and flow velocity fields inside the DIANA facility were measured. The temperature measurements differed significantly from the DNS simulation results obtained in the previous DHC work conducted at PSI. The scoping studies made with Ansys Fluent 14.0 software suggested that the thermal radiation heat transfer should be taken into account in the DHC simulations. Future work will contain more accurate Fluent simulations aiming to produce data that can be compared with the measurements. The measurements on the particle depletion in a differentially heated cavity with turbulent natural convective flows are currently on-going.

• Information related to the progress of the Phebus FP, ISTP, ARTIST and OECD/NEA STEM programmes was distributed.

2.5.5 Reactor vessel failures, vapour explosions and spent fuel pool accidents (VESPA)

VESPA (Reactor Vessel failures, vapour Explosions and Spent fuel Pool Accidents) project binds together three different fields of nuclear safety and severe accident management: Structural integrity, steam explosions and spent fuel pool accidents.

Specific goals in 2012

The applicability of a commercial finite element code (Abaqus) for modelling large deformations of a reactor pressure vessel at high temperature was studied. The first aim was to compare the results obtained with the commercial code with the specific code (PASULA) developed at VTT. The Abaqus code will be utilised to simulate a core melt scenario of a small scale pressure vessel that was experimentally tested in the Sandia OLHF-1 experiment.

The steam explosion loads for PWR and BWR cavities will be evaluated with TEXAS-V and MC3D codes. A target is to train a new code user and expert on vapour explosion area. This task had conjunction with SAFIR2014 SMASH project in structural effects of steam explosions, EU SARNET2 WP 7.1 project and currently on-going OECD/SERENA2 project. Furthermore, VESPA project provided the Finnish contribution to Task 8 of NKS project DECOSE (Debris Coolability and Steam Explosions) 2012-2014.

The third purpose of the project was to develop an analysis tool for studies of loss of coolant accidents in spent fuel pools with PANAMA and MELCOR codes. The objective of the PANAMA sub-task was to develop calculational methods to evaluate the heat-up of the spent fuel and surrounding structures in a hypothetical accident of loss of water coolant in water cooled interim storages. There is a question of very complicated heat transfer phenomena. Also the applicability of MELCOR code to spent fuel pool accidents was being assessed.
MELCOR was used for modelling several different spent fuel pool accident scenarios for a Nordic BWR.

**Deliverables in 2012**

- An axisymmetric FE-model was generated according to the geometrical details and temperature dependent material properties provided by the OLHF-1 experiment. The thermal loading was modelled using measured temperature data which was interpolated to the nodes. The main result of the current simulations was the vertical displacement of the lower head bottom, which was computed relatively well by Abaqus prior to the rupture. In conclusion, commercial finite element code Abaqus is suitable for modelling large deformation at high temperature. The results are highly dependent on the utilised material parameters that need to be verified by experimental test data.

- In November 2012 a visit was organised at the KTH facility in Stockholm. The goal of the visit was to learn the use of the code and discuss upcoming issues. From the calculations it was concluded that with the predetermined reference case parameters steam explosions can occur in the simulation. The maximum explosion was obtained in the CRGT Failure scenario with a melt jet diameter of 0.07 m. At the distance of 3.68 m from the centre of the cavity the maximum detected pressure was 257 bar and the maximum impulse was 51.5 kPas. An extrapolation method is still needed for evaluating the loads on cavity walls.

- During the project many different ways to make the computation efficient was studied. Validation at high temperatures is an essential subtask and the mesh density is also an important factor. A special mesh generator was developed to model the gap area and other parts of the racks and walls.

- The loss of pool cooling accidents at the spent fuel pools in the reactor hall of a Nordic BWR were studied using the MELCOR 1.8.6 code. Studies were carried out with several different calculation nodalizations. Other investigated variables were the total decay heat power of fuel assemblies in the pool, the initiator of the accident (loss of pool cooling or
loss of coolant from the pool), the LOCA leak elevation, the alignment of the re-flooding injection and the use of a lid on top of the pool. From the results it was observed that ensuring natural circulation of air in the fuel is essential in preventing fuel damages. In cases where the air flow is prevented (loss of pool cooling, wall LOCA or a lid on top of the pool), the fuel is damaged in nearly all the cases. Only with the pool decay power being below 2227 kW, the melt of the fuel was prevented.

**Figure 2.5.5.2 Maximum fuel temperatures in boil-off and bottom leakage scenarios, with and without a lid.**

### 2.6 Structural safety of reactor circuits

In 2012 the research area "Structural Safety of Reactor Circuits" consisted of seven projects: Environmental influence on cracking susceptibility and ageing of nuclear materials (ENVIS), Fracture assessment of reactor circuit (FAR), Monitoring of the structural integrity of materials and components in reactor circuit (MAKOMON), RI-ISI analyses and inspection reliability of piping systems (RAIPSYS), Advanced surveillance technique and embrittlement modelling (SURVIVE), Water chemistry and plant operating reliability (WAPA) and Fatigue affected by residual stresses, environment and thermal fluctuations (FRESH).

#### 2.6.1 Environmental influence on cracking susceptibility and ageing of nuclear materials (ENVIS)

The objective of the joint VTT – Aalto ENVIS project is to support safe operation of NPP’s through increased understanding of the influence of light water reactor environments on the ageing and environmentally-assisted cracking (EAC) susceptibility of nuclear reactor materials. To meet these goals, several tasks are pursued dealing with e.g. irradiation assisted cracking, thermal ageing, dynamic strain ageing of stainless steels and characterisation of austenitic nuclear materials. International co-operation is important as a tool to bring the latest knowledge to Finland. Knowledge transfer and continuous education will secure uninterrupted availability of high-quality expertise for ageing management. Development of new testing capabilities creates possibilities to stay in the forefront of research.
Specific goals in 2012

The ENVIS 2012 project has several goals dealing with the long-term behaviour of nuclear materials.

Understanding the role of deformation on initiation of environmentally assisted cracking (EAC) in austenitic materials was in 2012 increased by continuing super slow strain rate tests (SSSRT) for nickel-based weld metal specimens made of Alloys 182 and 82. Further knowledge of modern nickel-based alloys was gained from a Master of Science work at the Aalto University, in which totally eight nickel-based dissimilar metal weld (DMW) samples, received from EPRI, USA, were characterised. The behaviour of nickel-based materials is also the topic in the task dealing with Low Temperature Crack Propagation (LTCP) susceptibility. In 2012 fracture toughness investigations on Alloys 182, 152 and 52 was continued with special emphasis on the role of pre-exposure to high temperature PWR water.

Thermal ageing of austenitic weld metals with a ferritic-austenitic microstructure is an area where rather little data exist so far. It is well known, that the degradation of mechanical properties is due to microstructural changes in the ferrite phase and that good correlations can be obtained between the mechanical properties and the ferrite hardness. However, the ferrite phase in austenitic weld materials is too finely distributed for conventional hardness measurements. To gain more data on this issue, thermally aged duplex stainless steel materials, received from KAPL, USA, with a composition and ferrite distribution close to those of nuclear welds were investigated using instrumented indentation with small forces, i.e., nanoindentation technique.

The effect of thermal ageing on Co-based hard-facing alloys, like Stellite 6, is an unexplored area. Thermodynamics predicts thermal ageing to be possible over very long LWR operation times, but the kinetics is unknown. If thermal ageing cause embrittlement of the material and degradation of the mechanical properties, it may create a need for replacement of these alloys in NPPs. Co-based hard-facing alloy Stellite 6 from a main steam gate valve guidance piston, which had been in operation about 30 years before cracking was observed, was investigated during 2012 using optical microscopy, SEM, EBSD, X-ray analysis and fracture mechanical tests.

Neutron irradiation has an influence on the microstructure and susceptibility to irradiation assisted cracking, IASCC. VTT contributes to collating data by performing FEGTEM characterisation on materials used in the Halden reactor for different experiments. In 2012, FEGTEM was used to examine irradiation-induced damage in a cold worked AISI 316 baffle bolt that was removed from a French nuclear power plant. The analysed section of the bolt had accumulated a dose of approximately 15 dpa.

IASCC has also been the topic of further investigations of a failed core basket from a VVER 400 plant bolt, made of solution annealed Ti-stabilised stainless steel, with a rather low dose of 2.9 dpa. Bolt cracking has occurred also in French PWRs, where the material is cold deformed stainless steel. Based on extensive assessments a threshold dose for IASCC of 4 dpa has been proposed in France. The failure analysis of the failed VVER 440 bolt revealed two cracks, one at the bolt shoulder, and one at the mid-length. The washer below this particular bolt had been unintentionally spot welded to the shielding plate. To gain more detailed understanding of the reason(s) for the cracking, the bolt material was further characterised in ENVIS 2012 using FEGTEM and finite-element (FE) modelling, which was performed to evaluate the stress distribution in the bolt.
Development of new test capabilities is an important part of the renewal of infrastructure such as the Centre for Nuclear Safety. In addition to renewal of infrastructure, also ensuring uninterrupted availability of high quality experts in the field of nuclear materials is essential. This is ascertained through continuous knowledge transfer and active international cooperation.

**Deliverables in 2012**

- In the performed SSSRT tests, Alloy 182 was found to be much more prone to EAC crack initiation compared to Alloy 82, in which only tiny cracks were observed after straining to 5% plastic strain at a strain rate of $1 \cdot 10^{-8}$ s$^{-1}$ in simulated PWR environment. The results are in agreement with published data.

- The Master of Science work containing detailed results from characterisation of DMWs was completed and published in 2012. The samples involved Alloy 690 as base metal and Inconel 52, 152 and 52M as filler metals. Differences in the behaviour of the filler metals were observed. Higher hardness was found in Inconel 52M, followed by Inconel 152 and 52, respectively. Inconel 152 showed different behaviour than Inconel 52 concerning carbon migration. Hardness peaks were observed at the Alloy 690 – 52 fusion lines and differences were observed in the microstructures of Alloy 690, depending on the product form. Extensive carbide banding was observed in hot-rolled and forged plates, while extruded Alloy 690 showed no carbide or grain size banding. Alloy 690 HAZ showed hardness increased due to increasing residual strains towards the weld interface.

- The LTCP tests on Alloy 182 and 152 showed about 30% higher fracture toughness values after 30 days pre-exposure to hydrogenated, high temperature water followed by fracture toughness testing at the known maximum LTCP susceptible temperature of 55°C, compared to values obtained without pre-exposure. The influence of the environment on the fracture toughness of Alloy 52 is small, and remains high with or without pre-exposure.

- The nanoindentation measurements on the two duplex alloys thermally aged up to 2000 hours at 427 °C were performed as a pattern and the data hitting respective phases were recorded to compile the information in Figure 2.6.11, showing much less hardness increase of the ferrite phase in the 2101 steel with a Cr$_{eq}$ of 23.6 than in the 2209 steel with a Cr$_{eq}$ of 26.0. This implies that steel 21010 can be used for significantly longer times than the reference weld metal before significant embrittlement occurs. The nanoindentation is also shown to be a viable technique for thermal ageing investigations.

![Figure 2.6.1.1. Instrumented indentation hardness (HIT, MPa) results as a function of exposure time (h) at 427 °C.](image-url)
The microstructural and X-ray investigations of the 30 year aged hard-facing alloy Stellite 6 revealed a microstructure containing indications of sigma phase, Figure 2.6.1.2, which is known to be brittle. The mechanical tests using fracture toughness and impact testing showed that the instability fracture toughness of aged Stellite 6 hard-facing is 38...42 MPa√m. In dynamic fracture toughness test the maximum load was as low as 319 N. This suggests that fracture toughness value under impact loading is clearly lower than in quasistatic loading. The investigations will continue in 2013 on as-welded and reference material, and all results will be published in the 16th environmental degradation conference.

Figure 2.6.1.2. (a) XRD and (b) fracture toughness results of aged Stellite 6 material.

The ATEM investigations of the bolt slice received from Halden with a dose of 15 dpa showed that the material had a distinct density of gamma prime precipitates with orientation relationship with the fcc lattice. High-resolution EDS-examinations with a nominal probe size of ~1 nm showed that the grain boundaries (GBs) had experienced serious effects of radiation induced segregation accompanied by precipitation. Maximum RIS values for Ni and Si enrichment and Cr depletion at the GB were 23.4 wt% Ni, 6.3 wt% Si, and 13.9 wt% Cr. Void surfaces were also analysed by EDS, and they showed a clear Ni and Si enrichment and Cr depletion. By using dark field imaging, some of the precipitates were observed to be located at the void walls. Hexagonal features were observed around some of the precipitates, while the voids tended to shrink upon EDS analysis.

The FEGTEM investigations of the failed Ti-stabilised stainless steel bolt with a dose of 2.9 dpa showed that the radiation induced grain boundary segregation (RIS) of chromium was similar to published trend curves, Figure 2.6.1.3a, while the amount of Si-segregation was higher. The amount of irradiation-induced dislocation loops was in line with the literature and the measured hardness values of the bolt. It is known, that molybdenum disulphide containing grease, which decomposes and creates an environment which increases the risk for SCC, may have been used during installation of the bolts. The FEGTEM investigations showed no signs of sulphur, nor any other harmful elements, which could have increased the SCC susceptibility. Thus, the role of the environment is not considered to abnormal nor being the main cause for the cracking. The FE modelling, Figure 2.6.1.3b, showed high stresses at the bolt shoulder and along the threads up to the mid-length of the bolt. The high stresses along the bolt length are a consequence of the spot weld, as this eliminates the clearance between the bolt and the shielding plate and thus cause stresses on the bolt from the thermal expansion of the shielding plate during start-up. These results are in very good correlation with the observations of the failure analysis. The exceptionally high stresses are considered to be the main cause for the
IASCC in the bolt, and the case is a good manifestation that IASCC, as IGSCC, is a combination of material, environment and stress. A journal article summarising all results has been submitted for publication.

(a) 
(b) 
Figure 2.6.1.3. (a) Grain boundary segregation and (b) FE-model of the bolt with a dose of 2.9 dpa, where the washer had been spot welded to the shielding plate.

- A biaxial creep testing device for fuel clad materials has been designed in 2012, and manufacturing of the parts is in progress. The device is based on bellows loading technique and special emphasis has been put to make the design multipurpose and suitable also for irradiated materials.

- An advanced course on nuclear materials, dealing with nuclear materials manufacturing, ageing and degradation issues, was given in 2012 with prominent international and national lecturers and about 50 attendees including all YG persons from the ENVIS project team. The lecture material is available at Aalto University as are the final seminar papers in the form of a publication. The international cooperation consists of participation in the Degradation and Ageing Programme (CODAP) project, which is a NEA project in which a knowledge base is established and information is collected on various degradation modes, starting with flow assisted corrosion. ENVIS 2012 also participated in the EPRI MRP Alloy 690/52/152 Expert Panel which deals with modern nickel based materials (used e.g. in OL3) and their behaviour, and where results from e.g. the ENVIS project are presented. Travel reports are written from all international meetings and delivered to the Reference Group 6.

2.6.2 Fracture assessment of reactor circuit (FAR)

The objective of the 4-year project is to develop and to validate numerical and experimental methods for reliable reactor circuit structural integrity assessment. Especially, to

- Develop and evaluate numerical structural integrity assessment methods; evaluate and further develop advanced methods - such as crack growth dependent submodelling technique and extended finite element method (XFEM) - for nuclear structural integrity assessment. Evaluate uncertainties and conservatisms of fracture mechanical structural analysis methods. Applicability of more simple engineering structural assessment tools and numerical software will be studied.

- Study applicability and limitations of leak-before-break (LBB) approach; Limiting factors, requirements for material input data, tearing instability, effect of
ageing, special features of LBB for narrow gap welds and dissimilar metal welds (DMW) will be considered. Experimental and analytical methods for LBB will be developed.

- **Evaluate growth and criticality of real cracks with shallow or irregular shape in structures; low-constraint fracture mechanical testing methods, transferability to structures and advanced numerical methods will be developed**

- **Develop structural integrity assessment procedure for dissimilar metal weld (DMW) based on realistic failure criteria of DMWs, and develop practice with which the zones that are most critical for fracture can be identified and their fracture toughness and mechanical strength can be determined reliably**

**Specific goals in 2012**

Engineering assessment methods subproject aims at implementation of promising extended finite element method (XFEM) in commercial FE-code Abaqus. Evaluating applicability of simplified fracture and numerical tools in nuclear applications will also be performed. In 2012, a procedure that introduces (almost) arbitrary cracks in uncracked component geometry FE-meshes using submodelling will be developed and implemented. The procedure will increase computational efficiency and reduce manual labour when carrying out large number of fracture analyses and provides means to perform sensitivity analyses with respect to crack geometries, loading and material parameters for complex cracked geometries where handbook solutions are not applicable. Implementation will utilize Abaqus input file format and solver but the methodology can be implemented into any other FE-software. The procedure can be later developed to model crack growth, e.g. under fatigue loading, or utilized when carrying out XFEM fracture analysis.

The LBB subproject aims at reviewing and applying LBB approach, development of testing capabilities for LBB input fracture mechanical data, understanding especially long crack fracture behaviour and preparing preliminary test and assessment procedures for narrow gap weld for next year. Long crack instability will be further studied in relation to mixed mode loading conditions. In connection to that, numerical analysis of the fracture behaviour of the test specimens will be made. Straight pipe FEM model will be created. Parametrical studies with various material properties (low fracture resistance) will be performed to define limiting fracture resistance in the worst case i.e. break before leak.

Low constraint fracture subproject aims at getting quantitative tools for evaluating criticality and growth of low constraint cracks. The results are applicable to surface cracks and real irregular crack shapes in nuclear structures. The constraint and biaxial effect and their interaction in combination with the effect of warm pre-stressing (WPS) will be investigated. FEM-model of a WPS case will be made.

The subproject on Dissimilar metal welds DMW aims at defining the existing gaps for the structural integrity analysis of DMW. Also, procedures for fracture mechanical testing which could indentify the most critical zone of DMW for fracture will be reviewed and developed. Role of the residual stresses in the DMW integrity analysis will be further clarified. Applicability and requirements of XFEM for DMW integrity assessment will be studied. Fracture mechanical and mechanical testing methods used in characterisation of DMW will be applied for a Ni-based (J-groove or plate) DMW received from EPRI. Microstructural characterisation of the DMW's will be made in ENVIS project.
Deliverables in 2012

- A recent advancement in numerical fracture mechanics is the extended finite element method (XFEM) that allows defining mesh-independent cracks in the analysis model. Popular FE-software package Abaqus has included an XFEM implementation in its more recent versions. Although the implementation has some limitations, generally it is currently the easiest way of carrying out XFEM fracture analyses.

The applicability of the XFEM implementation in Abaqus was studied earlier in the project. A very comprehensive overview on the same topic is given in a recent M.Sc. thesis by Leven and Rickert. They obtained a good stress intensity factor agreement for benchmark cases with errors less than 10 per cent. The reported drawback was that the acceptable accuracy requires very dense mesh in the crack tip. The proposed element size is from 0.8 per cent to 13 per cent, depending on desired accuracy. The analysis models utilized in the thesis had several hundred thousand elements most of which located at the crack region. Also, best accuracy was obtained when the mesh was designed to coincide with the crack boundaries. This is not practical if the crack location needs to be varied or is unknown. It was found in both studies that the mesh in the crack location most affects the accuracy of the stress intensity factors. However, the verification cases in Abaqus manuals utilize relatively coarse mesh indicating that good results can be obtained also with a coarse mesh if the crack is placed correctly with respect to element edges.

When the location of the flaw is not explicitly known or the critical areas need to be determined for design purposes building traditional FE-models with cracks is extremely laborious. The XFEM approach simplifies this task notably. Only one uncracked mesh of the component needs to be built and all desired crack locations, shapes and sizes can be analysed with the built mesh. The automated crack generation procedure introduced in this work allows the user to generate multiple analysis models by defining the location and size of the crack.

The procedure was demonstrated for simple cases in this report to show what kind of results can be obtained using a single FE-model and multiple analysis runs. The procedure presented in this work is currently implemented for elliptical axial and circumferential surface cracks for demonstrative purposes. The procedure can later be developed for crack growth modelling by creating a propagated crack based on the results obtained from an earlier analysis and a crack growth law such as the Paris law as was manually demonstrated in this work.

The agreement with the obtained stress intensity factors with reference results was reasonable at best. The results demonstrate the findings of Leven and Rickert. Accurate evaluation of stress intensity factors requires very dense mesh in the crack region. The developed procedure is usable with any element size but carrying out multiple XFEM analysis with a very dense mesh would take weeks of computing time. It is recommended that the critical locations are first screened using a coarse mesh and then selected cases are analysed with a denser mesh.

Utilizing submodels for defining a dense mesh around the crack region provides an attractive and flexible option for modelling cracks in various locations. This approach will be studied in the next phase of this work.

- The LBB concept was applied to two case studies. The first case was a circumferential crack in BWR piping having a force/moment or displacement/rotation loading. In the
force/moment load case the requirements of the LBB concept were not fulfilled, but in the displacement/rotation load case the requirements of the LBB concept were fulfilled.

The second computed case, a straight pipe, showed a similar behaviour as the pipe bend. The condition in a real pipeline is probably somewhere between the force and displacement load case and should be thus evaluated case by case basis using the actual pipeline and boundary conditions. The SQUIRT2 code for the leak rate evaluation can be successfully run even in the Windows 7 (or higher) environment with a dos emulator. The challenges to fulfil the LBB concept are related to rather high safety margins in the LBB concept. One is related to leak rate detection (safety margin 10) and the others to the crack size.

• The definition and results of the numerical modelling on Case 1 analysed by VTT in the CABINET project are described. The first of the two accomplished experimental loading cases included in the benchmark study is designated Case 1. The material is irradiated NiCrMo1 UP (weld metal) and the specimen type is WOL-100X. The experimental data used in the analysis were provided for the project by Areva NP GmbH.

The results of the finite element model and experimental work are in agreement. The model is somewhat “stiffer” in terms of the CMOD-load response. The ”stiffness” of the model also affects the estimated $K_{JC}$ values. The likely reasons are that the boundary conditions are simplified and a full contact analysis has not been carried out, and also, the crack front is modelled as a straight line and the actual elliptical geometry is not included.

• The localisation of the asymmetrical deformation typical to dissimilar metal welds (DMWs) and the presence of an uneven crack front influence the crack growth in a manner that often differs from that experienced in similar-metal welds. Metallurgical discontinuity also causes a complex loading and stress-strain state, which may cause residual stress and stresses due to thermal loading. When assessing the structural integrity of a component, both the loading and the load-carrying capacity are determined. The WRSs are included in the analysis on either the loading or capacity side, depending on the design strategy.

The components of primary interest in NPPs often have a bi-metallic or dissimilar metal structure, where the mechanical properties of the joined materials are different, and thus add to the formation of the WRSs. According to earlier experimental measurements, FEM analysis results and WRS recommendations, the WRS values are typically relatively high in NPP component welds which are in as-welded state, typically of the scale of yield strength in tension in and near inner weld surface. Earlier work has also demonstrated that, unlike similar metals welds, residual stresses in DMWs can promote also ductile fracture and therefore the influence of WRS can be neglected only in the case that the material fracture resistance is sufficiently high (simultaneously accounting for the mismatch effects).

The current WRS recommendations in the most commonly applied procedures: the ASME recommendations, the British Standard BS 7910: 1999, the R6 Method, Revision 4, the SSM Handbook, the SINTAP Procedure, the API 579 Procedure and the FITNET Procedure, are based both on the available experimental data and FEM analysis results. These published WRS data appears to have substantial scatter. In comparison to the as-welded condition, the PWHT appears to remarkably decrease the WRS level of the piping welds; the same applies to repair welds subjected to the PWHT.
Of the covered WRS recommendations, only the SSM Handbook provides WRS distributions also for a selection of real dissimilar (i.e. bi-metallic) weld types. In the case of the other covered WRS recommendations, the applied distributions are calculated applying different formulae derived separately for the ferritic steel and the austenitic steel. Further validation of the SSM Handbook WRS distribution solutions for DMWs of various metallurgical concepts and flaw locations & orientations will therefore be of utmost importance in the future, both by using experimental work and numerical computations, coupled with modelling of the actual welding sequence and its thermal history ’pass-by-pass’.

A very recent programme compared different WRS predictions particularly for a dissimilar (DMW) pipe weld with careful measurements on a mock-up weld. It revealed that careful modelling of material hardening and heat source description resulted in very good agreement with the experimentally measured WRS profiles. This is in line with the approaches utilising ‘pass-by-pass’ modelling in describing the weld thermal history as a part of the WRS distribution analysis. The comparison of handbook WRS profiles with the best estimate FE predictions using improved modelling showed that there is a need for updating the recommended WRS profiles in the original SSM Handbook. Consequently, further research was deemed necessary to validate predictions of other WRS geometries, especially weld repairs which can have a substantial influence on the WRS distribution in piping components.

Recent attempts to measure WRS distributions in DMWs using several experimental techniques have revealed considerable scatter among the results given by different techniques. Additional difficulties accrue from the gradient of very narrow local regions with significantly different microstructures and mecahnical & physical properties, including the weld interface. This poses serious challenges to further development of the techniques themselves, as well as to interpretation of the outcome of the measurements. In case of DMWs, knowing the inherent stress-strain properties of the different microstructural regions of the weld is therefore of utmost importance for the realistic and reliable structural integrity analysis taking into account of the welding residual stresses.

2.6.3 Monitoring of the structural integrity of materials and components in reactor circuit (MAKOMON)

Non-destructive testing (NDT) techniques are used to monitor the condition of the structures of reactor circuit during the operation of nuclear power plants. The in-service inspections (ISI) are normally performed during the short revision period. It is necessary to develop inspection techniques that can be applied to reactor circuit components where the access is restricted and therefore decreasing the reliability of inspection.

The objective of this project is to develop more reliable and more efficient ways to use non-destructive testing techniques for monitoring the structural integrity of the primary circuit components. Aim is also to verify the reliability of some NDT simulations. The main methods used in the MAKOMON project are different ultrasonic applications and ultrasonic simulation, eddy current method and radiography.

Specific goals in 2012

MAKOMON project 2012 consisted of five subtasks, each of them had their own specific goals. One task compares the indications received from different types of defects in ultrasonic evaluation. Understanding of different indications in ultrasonic inspections is important for
example in in-service inspections. It was found in the same project a year earlier that the commonly used EDM notches produced significantly stronger indications than the same size fatigue cracks. In the latest project the comparison continued with different fatigue cracks.

In the Figure 2.6.3.1 test results for phased array ultrasonic method with transverse wave for two cracks (mechanical and thermal fatigue) are presented.

![Figure 2.6.3.1. Phased array ultrasonic measurements on the thermal fatigue crack on the right and the mechanical fatigue crack on the left.](image)

Ultrasonic simulation can be used to optimize the inspection techniques, and to see the restrictions of the inspection. The simulation of the defects is more and more important in e.g. certification processes. Computer simulations allow hundreds of defects to be generated and responses to be calculated with a relatively small effort and therefore help the selection of the correct defects. Probability of detection (POD) curves provide valuable data for the reliability of the used NDT method. The aim in the project is also to calculate POD curves from simulated defects. A novel method laser ultrasonics is reviewed for new possibilities of detecting defects without any contact on the material surface.

In many cases the growing defects are located at the same place as the magnetite deposits in the steam generator tubing. For this reason it is more and more important to locate and size the piles of magnetite on the tubing. Magnetite on the steam generator tubing is usually well detected and to certain limit, the thickness of the magnetite can be measured. Development of eddy current method to measure the thickness of the magnetite pile between the steam generation tubes is an area of interest that continues in this project.
The potentials for replacing X-ray film with digital detectors in nuclear environment have been studied in the project. The research is concentrated on comparing the film technique to digital radiography in the aspect of the nuclear industry.

**Deliverables in 2012**

- Both thermal and mechanical fatigue samples are inspected using conventional and phased array ultrasonic equipment. In data analysis, the signal-to-noise ratio of all defect indications has been defined. The initial inspection results of the studies will be confirmed after cracks are opened and investigated.

- An ultrasound simulation model have been developed which will be used to generate POD-curves for ultrasonic testing. During 2012, initial measurements were done to verify the model against measurements with EDM notches and fatigue cracks. Results show that more development for the simulation is still needed, in order to use simulations to generate realistic POD-curves.

- An introduction report to the state of the art of laser ultrasonics has been written. Laser ultrasonic is a method that needs development work but is potential for the applications that need noncontact inspection.

- The measuring of the thickness of magnetite piles between two tubes simulating the horizontal steam generator tubes have continues with a new setup. Two probe method transmits the signal from other tube and receive the signal in the other adjacent tube.

- A report that compares digital radiography to the film technique has been written. The comparison has been made with the field of the nuclear power generation industry in mind. New project member has familiarized himself with radiography along this work.

- Education of the research scientists continued, including also courses of specific non-destructive evaluation methods. One SFS-EN 473 NDT level 2 certificates have been reached in 2012 (Jonne Haapalainen UT).

- Four research institute reports have been completed.

- Six persons from the project have attempted to three worldwide conferences of NDT.
2.6.4 RI- ISI analyses and inspection reliability of piping systems (RAIPSYS)

Risk-informed in-service inspections (RI-ISI) aim at rational in-service inspection management by taking into account the results of plant specific risk analyses in defining the inspection program. The fundamental idea is to identify risk significant locations where the inspection efforts should be concentrated. Even though RI-ISI has been widely applied in the U.S., European utilities and safety authorities consider that several issues need further research, and that the U.S. approaches cannot be adopted as such. In Finland, the implementation of RI-ISI is a topical issue. RI-ISI is a rather resource-demanding process, and depends on the level of detail of various analysis parts. It is necessary to show through research studies what simplifications can be justified, in order to have a robust and reasonable methodological approach. This project is a continuation to the corresponding RI-ISI related project in the previous SAFIR program, i.e. SAFIR2010. The focus is on the remaining open questions and further development issues as well as providing guidance on the application of a RI-ISI program.

Specific goals in 2012

The project was organised in three subprojects, and the specific goals of each subproject are described below.

In the first subproject, the goals were related to further development of probabilistic analysis methods for estimating pipe component failures. More specifically, this concerns expansion and improvement of probabilistic computation procedures and capabilities of analysis code VTTBESIT as well as more accurately taking into account loads and reliability of piping system inspections. Another issue is to improve the crack propagation computation procedure of probabilistic VTTBESIT, e.g. by developing the crack growth increment computation part of the code. In connection to this, it is planned to examine whether probabilistic distributions could be developed for the material type and environment specific parameters used in the associated crack growth rate equations. The aim is also to work on a dissertation concerning structural reliability methods and RI-ISI, so as to complete it within the duration of the research program.

The goals of the second subproject were related to risk ranking, selection of inspection sites and acceptance criteria of a RI-ISI program. An important part of this was the participation in the activities of the European Network for Inspection and Qualification (ENIQ) Task Group Risk (TGR). ENIQ TGR develops recommended practices and discussion documents related to RI-ISI. As a new issue, the benefits and applicability of risk informed safety margin characterisation (RISMC) approach to RI-ISI analyses are examined.

Another task of the second subproject was to examine the effect of initial flaw and load assumptions as well as of inspections on risk estimate changes, see Figure 2.6.4.1 for an example concerning computational results. The duration of this task is planned to be three years, and during the second year the aim was to perform the main part of the computational analyses as well as to provide preliminary/tentative conclusions. Further, the aim was to send an article manuscript to a peer reviewed scientific journal on the probabilistic assessment of sizes of initial cracks caused by stress corrosion cracking (SCC) under BWR conditions.

The third subproject covers other international co-operation (besides participation to ENIQ TGR activities) and project management. As for the former issue, the aim was mainly to participate in RI-ISI related conferences.
Figure 2.6.4.1. For two different inspection intervals (II) and three qualities, the leak probability for one BWR primary circuit pipe weld over 60 years, when the assumed degradation mechanism is stress corrosion cracking and weld residual stresses are according to ASME recommendations.

Deliverables in 2012

- The results of the research work on structural reliability analysis methods focus on further development of a more accurate and versatile approach for probabilistic crack growth analysis tool VTTBESIT. The purpose of these efforts was to improve the probabilistic degradation computation capability needed in the degradation potential assessments concerning the RI-ISI analyses.

- The study on the material type and environment specific parameters used in the crack growth rate equations resulted in collection of relevant parameter data as well as on examination whether probabilistic distributions could be developed for these parameters. The considered environments were primary circuits in BWR and PWR plants.

- The VTT contributions to ENIQ TGR documents included those on "ENIQ TGR discussion document: RI-ISI – Lessons Learned from Application to European Nuclear Power Plants" This document will be published as ENIQ report No. 48.

- The results from the second part of the study on the effect of initial flaw and load assumptions as well as of inspections on risk estimate changes concern extensive failure potential and risk analyses with probabilistic VTTBESIT code and Markov process based application for three representative NPP piping welds, covering a wide range of initial flaw and load assumptions as well as inspection strategies. The Markov process based risk computation application was further developed by expanding its selection of probability of detection (POD) models.

- A scientific journal article “A study on the effect of flaw detection probability assumptions on risk reduction achieved by non-destructive inspection” was published in September 2012 in Reliability Engineering and System Safety (RESS).
• Concerning a study on the applicability of the NPP piping degradation databases for estimation of crack initiation and leak frequencies in NPP primary circuit components, participation with presentation and article “On applications concerning OECD pipe failure database OPDE” in the Annual European Safety and Reliability Conference (PSAM11-ESREL2012) in June 2012.

2.6.5 Advanced surveillance-techniques and embrittlement modelling (SURVIVE)

The SURVIVE project has three separate tasks, namely 1) Validation of small specimen test techniques, 2) Development of multiscale modelling of fracture and 3) Irradiation induced microstructure and embrittlement modelling. Main focus of the works is on the behaviour of relatively high phosphorus VVER440 welds, which are currently utilized in NPP’s in Finland (2 units), Russia (4 units), Ukraina (2 units) and Armenia (1 unit) and which materials are not studied in European Community programmes. Small specimens are utilized widely in irradiation effect studies but validation of the methods has not been always thoroughly performed. In multiscale modelling of cleavage fracture the multicristalline nature of steels is introduced instead of assuming quasi-homogeneous materials. In the microstructural characterisation work physical bases of embrittlement is aimed to be identified. The work is performed in co-operation with foreign partners in order to utilize the best available experimental facilities. Own work is focussed on resistivity measurements.

Focus of works in 2012

1) Validation of small specimen test technique

Validation of specimen reconstitution for VTT’s electron beam welding technique was largely completed during 2011. During 2012 residual stress measurements as well as the related specimen bending measurements were performed, which completes the sub-task.

The main topic in small specimen test technique is currently validation of a miniature CT-specimen for Master Curve $T_0$ measurements, which is performed within a CRIEPI/Japan joint research programme. VTT signed the formal agreement with CRIEPI during 2012, received the ready-made specimens and a miniature clip-gauge and testing was started. The other participants are JAEA, Kyoto University, Toshiba/Hitachi-GE, MHI, UJV, ORNL and EPRI. The programme offers a good contact forum and an opportunity to demonstrate VTT’s capabilities.

As a VTT own research effort experimental comparison of $T_0$ values measured with 3PB- and mini CT-specimens was initiated. Mini CT-specimens defined by CRIEPI and 3PB-specimens having identical dimensions (B,W) will be used. Relatively large data base will be created in order to allow reliable comparisons.

VTT participated in the NIST/USA organized round robin experiment on instrumented impact tests performed with miniature Charpy-V (KLST) specimens. The data has been forwarded to NIST. Other participants are IAM (EU), SCK-CEN, UJV, CIEMAT, ORNL and HZDR, i.e a good contact surface for VTT.

2) Development of multiscale modelling of fracture

Crystal plasticity analyses have been carried for microstructural aggregate models of pressure vessel steel. The aggregates were generated to display the differences in sub-grain
deformation behaviour in different microstructural morphologies. The results demonstrate the effects, which anisotropic plastic deformation has in stress state within microstructure, and as such the findings can be used to improve computation of cleavage initiation probability.

3) Irradiation induced microstructure and embrittlement modelling.

The main focus of microstructural studies was on APT data measured by Tohoku University for weld 501, which was used as surveillance weld for Loviisa-1 vessel annealing. Data is now available for the unirradiated-, I-, IA-, IA1-, IA12- and IA13-conditions. The data indicates that small copper clusters including some phosphorus are formed during irradiation, some of them grow bigger during annealing but others dissolve back to iron matrix. During re-irradiation small clusters start to form again because oversaturated phosphorus remains in the iron matrix. Stable composition of the precipitates formed during re-irradiation is the same as composition of the ones formed during initial irradiation. The large copper clusters formed during annealing do not contain phosphorus. Phosphorus is found also on many other locations like on vanadium carbide and in dislocations. The hardening effect of copper rich formations estimated by Russel-Brown is not able to explain material hardening during re-irradiation. The observed solid clusters in the re-irradiated condition are shown in Figure 2.6.5.1 and the matrix content of copper in Figure 2.6.5.2.

![Solid clusters formed in weld 501 during re-irradiation after 1, 2 and 3 re-irradiation cycles in Loviisa surveillance channel.](image)

Figure 2.6.5.1. Solid clusters formed in weld 501 during re-irradiation after 1, 2 and 3 re-irradiation cycles in Loviisa surveillance channel.
Figure 2.6.5.2. Copper content of iron matrix after different irradiation-annealing-reirradiation cycles. 270°C is the operation temperature of the reactor and 475°C is the annealing temperature of the specimens. When the matrix is oversaturated by copper, copper rich solid clusters may precipitate. The matrix is all the time saturated by copper. Annealing increases the copper content of the matrix considerably i.e. some formations dissolve back into the matrix even if other solid clusters grow bigger at the same time.

It was agreed in principle that CRIEPI/Japan will perform APT measurements with four different weld materials available in the unirradiated-, I-, IA- and IAI-conditions. The material originates from the EDF-financed research programme, where RRC-KI, Fortum Ltd and VTT participated. The samples will be prepared by VTT and delivered to Japan.

No embrittlement modelling work was performed during 2012.

2.6.6 Water chemistry and plant operating reliability (WAPA)

The overall objective in the project “Water chemistry and plant operating reliability” is to study the role of water chemistry in preventing degradation of the components both in the primary and secondary side of NPPs.

In BWR reactors and PWR primary side the main target is to optimise the water chemistry with regard to mitigation of activity build-up by minimising the formation of corrosion products and to mitigate stress corrosion cracking. In PWR secondary side the main target is to optimise water chemistry for minimising the formation of colloidal magnetite and to minimise the deposition of magnetite onto steam generator surfaces.

Specific goals in 2012

Specific goals in 2012 were to find an optimal water chemistry for the pre-passivation stage of the primary circuit of a new PWR reactor and to study the effect of water chemistry on the magnetite deposition in the PWR secondary circuit.
Optimisation of the pre-passivation water chemistry of a PWR new build

The primary circuit of a new PWR reactor is passivated (preoxidised) before loading the first fuel. This procedure is called Hot Conditioning which is part of the Hot Functional Testing (HFT). The main purpose of the preoxidation is to minimise the concentration of corrosion products in the coolant during future power cycles, and thus minimise corrosion damage and activity build-up at the plant. There is no international consensus on the best available procedure for HFT. Open questions exist on the minimum length of passivation time, the optimal concentration of Li, and the use of boric acid.

As part of the co-operation between VTT and Bhabha Atomic Research Centre (BARC) in India the pre-passivation in Pressurised Heavy Water Reactors (PHWRs) was first studied. For this purpose, the use of electrochemical impedance spectroscopy (EIS) as an acceptable on-line tool for following the progress of pre-passivation was verified by comparison with the conventional coupon exposure technique. The electric and transport properties of the oxide formed on carbon steel in simulated PHWR hot conditioning water chemistry change with the oxidation time and stabilise after ca. 30–40 h of exposure. This was in line with the observations at the RAPP-5 plant as well as the conventional coupon studies.

After verifying the applicability of the in situ EIS technique for determination of the quality of the oxide film forming on carbon steel during pre-passivation of a PHWR plant, the same approach was used to study the pre-passivation of Alloy 690 under simulated pressurised water reactor (PWR) hot functioning water chemistry conditions. Alloy 690 is typically used as steam generator (SG) tubing in PWR new builds. Steam generator tubing forms a major part of the surface area (70 %) exposed to the primary coolant in a PWR plant and thus governs the release of corrosion products into the primary circuit. Optimisation of the PWR hot conditioning water chemistry mainly aims at forming the most corrosion resistant oxide layer on the SG tubing.

The effect of lithium (as lithium hydroxide LiOH) on Alloy 690 passivation was studied in the range 1…2 ppm and that of boric acid (H$_3$BO$_3$) in the range 0…1200 ppm. Fig. 2.6.6.1 shows a compilation of the data for Alloy 690. Increasing lithium concentration is seen to decrease the corrosion resistance markedly. Addition of boric acid can counteract the detrimental effect of lithium to some extent. The optimal water chemistry for pre-passivation of a PWR new build, based on these results, would be Li = 0.5…1.0 and H$_3$BO$_3$ = 200…600 ppm. Here, the function of boric acid is both to buffer the pH elevating effect of lithium hydroxide and to aid in passivation of the surface of Alloy 690.

Studies on magnetite deposition in PWR secondary circuit

Magnetite ($\text{Fe}_3\text{O}_4$) is formed in the pressurised water reactor secondary circuit mainly from corrosion of carbon steel tubing and other carbon steel components. Magnetite particles originating from the corrosion process are transported with the flow through the feed water line and deposit e.g. in the flow holes/crevices between the steam generator tubes and tube support plate and on the tube sheet creating flow and corrosion problems.
In aqueous solutions, colloidal magnetite particles usually have a surface charge, the sign and magnitude of which depends on the water chemistry conditions (e.g. pH and chemicals used to adjust pH). Surface charge (i.e. zeta-potential) is the main variable in determining the ease of deposition, and it can be influenced by water chemistry additives (e.g. amines and dispersants). A new streaming potential experimental arrangement (in principle a small-size once-through feed water line simulator) was developed for the zeta-potential measurements and positively verified by comparing the results of zeta-potential measurements at room temperature with those gained with a commercial system. Increasing temperature was found to decrease the magnitude of the zeta-potential in water with ammonia and ethanolamine, but not in water with morpholine. Thus, increasing temperature tends to promote the deposition tendency of colloidal magnetite particles in PWR secondary side water conditioned with ammonia and ethanolamine.

**Deliverables in 2012**

- The work on optimisation of the water chemistry for pre-passivation of the primary circuit of a PWR new build was completed and the results were published as reports, a conference and a journal paper. According to the results, the water chemistry normally used in Japan is closest to optimal.

- The effects of hydrazine on carbon steel corrosion were studied as well as the effect of ammonia, morpholine and ethanolamine on the tendency for deposition of magnetite under PWR secondary circuit conditions. Two reports were published with preliminary
results. A new streaming potential experimental arrangement (in principle a small-size once-through feed water line simulator) was developed as part of an MSc-thesis work, and the thesis was published. Work in this area will continue in 2013.

- Stability of colloidal particles and deposit consolidation in nuclear power plants was studied as a literature work. This study is part of a larger effort aiming at development of a model describing dissolution, deposition and re-entrainment of magnetite and soluble iron species in the secondary circuit conditions.

2.6.7 Fatigue affected by residual stresses, environment and thermal fluctuations (FRESH).

The project aims at expanding and deepening understanding of fatigue behaviour experienced by nuclear power plant (NPP) pressure boundary components under realistic loads and environment. The accuracy of the fatigue analyses are planned to be significantly increased by:

- Determination of the realistic stresses in a weld prior to and during NPP operation,
- determination of the realistic loads caused by turbulent mixing and NPP environment, and
- clarification of stress component categorization for numerical fatigue analyses

Education of new experts is essential part of this research project. The simulation of the phenomena that are also studied experimentally will require understanding in both fields and will ensure that new research scientist will become experts in their field.

Specific goals in 2012

The project was organised in four subprojects, and the specific goals of each subproject are described below.

In the first subproject, the goal was determining the environmental effect to fatigue more realistically during NPP load transients. In reality, the experienced loads create a three dimensional stress and strain state in the pipe components. The FPIPE program and its ASME output handler use the methodology developed by FEMdata and TVO, and are validated by VTT. They compute the $F_{en}$ values for realistic NPP load transients by taking into account the multiaxial strain state. The goal was also to determine how the measured load transients differ from design transients in terms of fatigue and $F_{en}$-factor values. In addition, the variance in the $F_{en}$-factor and fatigue usage factor values was studied during different measured transients that are categorized as same load events.

One goal of the second subproject was the accurate determination of residual stresses in a multi-pass welded mock-up pipe by simulating the welding process with finite element method. Residual stresses will be measured with multiple methods and results will be reported in future. Computational simulation of welding will provide one more independent result for the residual distribution. Figure 2.6.7.1 shows computed circumferential residual stress after welding.
Figure 2.6.7.1. Computed circumferential residual stress (MPa) after welding.

Another goal of the second subproject was to determine the pros and contras of weld overlay as a long term repair method in the form of literature survey.

The third subproject aimed at collecting and reviewing the procedures concerning: categorization of thermal stresses and their interaction with other stress components, plastic strain correction factor, $K_e$, for taking into account elastic-plastic effects in the computation of stress ranges, and linearization of through wall stress distributions. When using FEM applications for solving stresses in piping components during NPP operation, the obtained stress distributions are a sum of different underlying factors, such as pressure, stress concentration, thermal transients as well as external forces and moments. As an additional challenge, in the RCC-M code and ASME Code Section VIII the $K_e$ factor is defined separately for mechanical and thermal stress components. Thus, this necessitates several FE computations separately under mechanical and thermal loads, in order to obtain the mechanical and thermal stresses. In this subproject, a state of the art review was performed on procedures concerning stress categorization, $K_e$ factor and stress linearization.

The goal of the fourth subproject was to model more realistically the thermal fatigue caused by turbulent mixing in a T-joint. The results were used for assessing the validity of the sinusoidal (SIN) method in different cases and for developing the method further. During the first year, detailed coupled CFD-FEM calculations carried out in EU FP7 project STYLE were continued. Fatigue calculations were performed with the CFD-FEM approach and with the SIN method.

Figure 2.6.7.2 shows validation of the CFD results against measured data in the T-joint. For the mean temperature, the both inlet conditions yield similar results and agreement with the experiment is good. For the root-mean-squared (rms) temperature, the turbulent inlets yield generally slightly lower values than the steady inlets. Agreement with the experiment is generally fairly good also for the rms temperature.
Figure 2.6.7.2. Normalized mean and rms temperature 1 mm from wall along the main pipe of the T-joint. (─ CFD steady inlets, — CFD turbulent inlets, ● Experiment)

**Deliverables in 2012**

- The report concerning the environmental effect contains fatigue and $F_{en}$-factor values computed with design load transients as well as measured load transients that represent the extreme cases from years 2001-2011. It was found that the measured load transients are mostly conservative even though the environmental effect is taken into account. All in all, both measured and design transients induced little fatigue at the selected locations. In addition, the study shows the strain range occurring during actual load events.

- The report concerning the weld residual stresses shows the simulated residual stresses in a multi-pass welded mock-up pipe. Residual stresses are being measured at Aalto University from the same mock-up that was used in the simulation. In the future, the simulated results can be compared to measured ones, once they are published.

- The report concerning the weld overlay procedure indicates that the weld overlay is an efficient method for mitigating inter granular stress corrosion cracking (IGSCC) in a NPP environment by providing an IGSCC resistant material barrier and by producing beneficial compressive stresses at the inner surface of the pipe. Numerous experimental, analytical studies and field experience support the beneficial features of weld overlay.

- The study on stress categorization, $K_e$ factor definitions and stress linearization summarises the different approaches and procedures provided by widely used structural integrity codes/guidelines as well as recent publications on the issue.

- The report on the simulations of thermal fatigue caused by turbulent mixing contains fatigue calculations with different modelling techniques. The CFD calculations were first validated against measured temperatures and velocities in the mixing Tee. Fatigue results obtained with the CFD-FEM and SIN methods were then compared. In addition, the effective heat transfer coefficient was determined from the CFD calculations.
2.7 Construction Safety

In 2012 the research area "Construction Safety" consisted of four projects: Impact 2014 (IMPACT2014), Aging management of concrete structures in nuclear power plants (MANAGE), Structural mechanics analyses of soft and hard impacts (SMASH) and Seismic safety of nuclear power plants - targets for research and education (SESA).

2.7.1 Impact 2014 (IMPACT2014)

A general objective of Impact 2014 project is to obtain experimental information on the physical phenomena involved in a condition where a passenger airplane impacts against a nuclear power plant. The missiles used in impact tests are describing wings, engines and carriages of aeroplane and also model of fuselage. Some tests have been done using water filled missiles to research the fuel spreading. The target used in the tests, has been force plate or concrete wall with reinforcement and liner. The wall has been designed with pre-stressing bars and liner and later on also curved structure or floor-wall structure will be used. The test results have also be used for numerical analysis in SMASH project in SAFIR 2014.

The impact project includes 9 partners from Europe, Canada and USA, which are funding the project and also analysing the test results of project. The specific goals, test matrix and test schedule of tests will be decided in TAG (Technical Advisory Group) meetings with Impact partners two times in a year.

Specific goals in 2012

1 Improving of test apparatus

The test apparatus will be modified to be suitable for liner, curved, and floor-wall structures in 2013. The renovation of Impact hall has decided to perform in 2013, but the new Impact apparatus capable to test concrete walls dimensions of 3.5 m* 3.5 m has been designed and will be manufactured in summer 2013. The pressure accumulator and the acceleration tube will be hoisted to the location of the new centre point of the wall by installing higher supporting frames under the tubes. The wider walls will be prepared and casted in the future in situ in hall and then moved and lifted to the testing position in front of the acceleration tube. The new wider wall has been designed and the uplifting winch device has been constructed. The new Impact hall will be closed by building new pressure walls and doors to close the hall. Also new monorail crane will be obtained and fixed on the ceiling of the hall.

2 Improvement of measuring system

The data acquisition system has been designed to be able to measure forces and stresses when floor-wall structure will be used as target. According to TAG meeting the first floor-wall structure will be tested in 2013. The permanent deflection of the wall has been measured by laser and deflection sensors before and after the test. New measuring device including 8 measuring channels and one fibre optical measuring card has been obtained for new test types. The permanent strains of reinforcement have been measured by optical fibres using static test and also in some impact tests, see Figure 2.7.1.1. The deceleration of missile has been measured by accelerometers and strain gauges fixed in missile itself. The measured data has been saved into the base memory of new wireless measuring device developed for impact tests.
3 Pre-calculation of the tests

All the tests made by stainless steel and wet missiles and concrete, liner, curved and floor-wall structures have been pre-calculated and designed in order to achieve the desired failure mode. Complex structures e.g. liner walls, curved walls and floor-wall structures have to be designed before testing to be able to have desired behaviour for the structure. The analysing and calculations have been done also by project partners and the also the post analysing in SMASH project of SAFIR program.

4 Testing of missiles, walls and structures

The main purpose of the test campaign in 2012 was to test concrete walls, thickness of 150 mm or 250 mm using stainless steel missiles or rigid steel missiles. Some tests have been done using water filled missile and the main idea was to measure the sizes and velocities of droplets. The test matrix is decided by TAG meeting and is given in Table 2.7.1.1. The titles for tests in the new international project of Impact 3 are (1) Punching behaviour (2) Bending behaviour (3) Combined bending and punching (4) Vibration damping (5) Liquid effects. According to table 2.7.1.1 in 2012 the following tests have been performed, one bending test with the slab thickness of 150 mm, 6 punching tests with the slab thickness of 250 mm, two combined bending and punching tests with the slab thickness of 250 mm, two static bending tests with the slabs thickness of 150 mm and width of 700 mm. In addition six force plate tests have been performed with totally or partly water filled with missiles.

5 Archiving of test results

All the drawings, places of sensors, material test results, videos, photos and measured data e.g. will be saved systematically to VTT DOHA system to be distributed to all partners. All the tests (missile and concrete structure tests) will be post-analysed to have quickly results from all the tests and to be able to correct tests specimens, apparatus or measuring system before next tests. The systematic result analysis will be done by LabView program. The data will be used for FE analysis in SAFIR 2014/SMASH project.

*Figure 2.7.1.1. Static test slab during the beginning (left) and after the test (right).*
Table 2.7.1.1. Tests conducted within IMPACT project in 2012 with main data.

<table>
<thead>
<tr>
<th>code</th>
<th>type</th>
<th>target</th>
<th>$v_0$ [m/s]</th>
<th>$m_m$ [kg]</th>
<th>missile material</th>
<th>$l_0$ [mm]</th>
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<tr>
<td>F1</td>
<td>bending</td>
<td>150mm, 6mm closed stirrups</td>
<td>144</td>
<td>50.2</td>
<td>st. steel 2mm</td>
<td>2361</td>
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<tr>
<td>P1</td>
<td>punching</td>
<td>250mm, 12mm hooked stirrups</td>
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<td>47.4</td>
<td>steel 12.5mm</td>
<td>640</td>
</tr>
<tr>
<td>P2</td>
<td>punching</td>
<td>250mm, prestressed and grouted tendons</td>
<td>123</td>
<td>47.5</td>
<td>steel 12.5mm</td>
<td>640</td>
</tr>
<tr>
<td>P3</td>
<td>punching</td>
<td>250mm, increased bending reinf.</td>
<td>140</td>
<td>47.5</td>
<td>steel 12.5mm</td>
<td>640</td>
</tr>
<tr>
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<td>punching</td>
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<td>120</td>
<td>47.5</td>
<td>steel 12.5mm</td>
<td>640</td>
</tr>
<tr>
<td>P5</td>
<td>punching</td>
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<td>steel 12.5mm</td>
<td>640</td>
</tr>
<tr>
<td>P6</td>
<td>punching</td>
<td>250mm, 10mm T-headed bars</td>
<td>111</td>
<td>47.5</td>
<td>steel 12.5mm</td>
<td>640</td>
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<td>combined</td>
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<td>st. steel 3mm</td>
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<td>st. steel 3mm</td>
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<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
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<td>static</td>
<td>150mm, narrow</td>
<td>-</td>
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<td>100</td>
<td>50/37</td>
<td>st.steel 1.5 mm</td>
<td>2050</td>
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<tr>
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<td>force plate</td>
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<td>50/37</td>
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<td>2120</td>
</tr>
<tr>
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<td>st.steel 1.5 mm</td>
<td>2050</td>
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</tr>
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<td>100</td>
<td>50</td>
<td>st.steel 1.5 mm</td>
<td>2050</td>
</tr>
</tbody>
</table>

**Deliverables in 2012**

**Improving of test apparatus**

- The new Impact apparatus, new supporting frame and new testing structures have been designed for the future tests with higher walls in new location.

**Improvement of measuring system**

- The measuring system has been improved and the number of measuring channels has been increased up to 32. The new equipment has been obtained to measure strains using fibre optics and permanent deflections have been measured by laser devices. The deceleration of missile has been measured by wireless equipment.

**Pre-calculation of the tests**

- All the tests including the new vibration/damping test have been pre-calculated to have successful tests.

**Testing of missiles, walls and structures**

- In 2012 totally 17 static of impact tests have been performed. Six tests have been done with water filled missile and 2 tests have been static tests.
- In 2012 one concrete wall, thickness of 150 mm has been tested to research bending effects.
- In 2012 six concrete walls, thickness of 250 mm have been tested to research punching effects.
In 2012 two concrete walls, thickness of 250 mm have been tested to research combined bending and punching effects.

Archiving of test results

All the tests have been archived to VTT DOHA data base. The results include measured signals, material tests results, drawings, photos and high speed videos. The data can be viewed and reported by standalone LabView program, made during the project.

2.7.2 Aging management of concrete structures in nuclear power plants (MANAGE)

The main objective of the MANAGE project is to develop a platform for the ageing management of concrete infrastructure at Finnish nuclear power plants. This platform supports many analysis and design tools related to the management of concrete structures. Using the same harmonised data system, applications related to the safety, performance, and service life management can be utilized. Several of the applications used in the analyses are developed during the project. By the ageing management system the acquisition of essential, up-to-date and proactive data on the condition and performance of concrete structures is secured.

A new inspection database will be developed. The inspection database is a relational database and will contain all observations from periodical inspections and special inspections. The aim is also to develop a monitoring and simulation system for the NPP concrete structures and an interface for structural analyses. The service life management system which was developed in SAFIR 2010/ SERVICEMAN project is linked to the platform and the new database system.

MANAGE is a joint project with VTT Technical Research Centre of Finland and Aalto University.

Figure 2.7.2.1. Schematic representation of the structure of the MANAGE platform and the user applications
Specific goals in 2012

The specific goals for the MANGE 2012 project were the following:

• Preliminary version of the MANAGE platform Graphic User Interface;
• Conclude physical design and implementation of the central database;
• Conclude physical design and implementation of the inspection database;
• Preliminary version of data visualisation application;
• Preliminary version of application for integrating the life cycle design software ServiceMan;
• Prepare harmonized and systematic classification of condition evaluation for the data management system;
• Non-Destructive evaluation of the pre-stressing tendons at Olkiluoto NPP;
• Participation in the activities of OECD NEA-CSNI-WGIAGE: Working Group on Integrity of Components and Structures (Integrity and Ageing of Concrete Structures), and
• Dissemination in the form of conference presentations and papers.

Deliverables in 2012

• An In-service Inspection and Inspection Data Registering of Concrete Structures in Nuclear Power Plant report was prepared (in Finnish and English) to help the inspection engineer prepare the inspection data for facilitated insertion into the inspection database of the MANAGE platform
• A report on the use of several Non-Destructive Testing equipment’s for the Condition testing of pre-stressing tendons at Olkiluoto 2 NPP was prepared by Ramboll Oy.

2.7.3 Structural mechanics analysis of soft and hard impacts (SMASH)

The main objective of this project is to develop and take in use numerical methods for predicting response of reinforced concrete structures to severe dynamic loads such as impacts of projectiles and pressure waves due to explosions. The structures may additionally be pre-stressed or covered with a steel liner. The aircrafts contain fuel which leads to a high risk of fire. The aim of the liquid research is the validated capability to simulate spreading of burning liquid and smoke as a result of an aircraft impact.

It is important to evaluate and prioritize the development needs based on the work already done. The post-calculation of the tests conducted in the ongoing IMPACT project gives valuable information on the applicability and development needs of the methods.
Specific goals in 2012

The main aim was to verify methods and models in assessing combined bending and punching of a wall loaded by a soft missile using FE method.

A user-defined subroutine to define concrete material behaviour was be implemented. This subroutine incorporates independent failure criteria for both tension and compression. The behaviour are strain rate dependent both in tension and compression. The cohesive surface methodology studied in case of explosions was applied and tested also in case of hard missile impacts. Cohesive surface methodology to model fracture of concrete was further developed.

Impacts due to explosions: State of the art on existing analysis methods and test results was compiled.

The goal of the liquid research is the validated capability to predict heat exposure and smoke spreading from a fuel release of aircraft impact.

Deliverables in 2012

• Post-analyses for selected IMPACT Tests: One impact test series was carried out by using as a target a two-way supported concrete plate with a wall thickness of 15 cm. The impact velocity in the eight tests was varied from 110 m/s to 160 m/s. Two types of missiles were used in the tests. Four tests were carried out using a stainless steel missile with a mass of 50 kg. The other four missiles with the same total mass of 50 kg were equipped with a water tank, containing 25 l of water, at the front of the projectile. These soft missile tests are post-calculated with finite element method (FEM) and simplified methods. The numerical results are in reasonable agreement with the measurements.

• The whole frame structure to which both the impacted slab and deflection sensors are attached is modelled with finite FEM in order to assess its effect on the displacements of both the slab and the sensors. According to this study, the frame has a minor effect on the maximum peak value and no effect on the permanent deflection values, but a more observable effect on the measured post-impact oscillation data.

• Updating the DETO code for the modelling of near-field blasts. (Fig. 2.7.3.1)

• A new simulation method called cohesive surface methodology was applied in simulating spalling and breaching of concrete walls due to shock waves. (Fig. 2.7.3.2)

• The liquid measurements were used in the plant-scale impact fire simulation examining the physical extent of the flame and smoke influence, as well as the times required from the flames and smoke to reach different parts of the NPP construction (Fig. 2.7.3.3).
Figure 2.7.3.1. Pressure loads applied to the slab top surface. The pressures dependent on both time and radius were defined by DETO code v4.0.

Figure 2.7.3.2. Final deformed shapes for all simulated cases. Contour color indicates magnitude of equivalent plastic strain with grey areas over 3 %.
2.7.4 Seismic safety of nuclear power plants – targets for research and education (SESA)

The decisions to increase the number of nuclear power plants (NPP) in Finland, and especially the positioning of one NPP in northern Finland calls for the need to assess the seismic hazard and the potential effect of earthquakes on plant safety requirements and design criteria for new installations. In order to address these questions SESA involves all aspects of the seismic assessment from: evaluation of the hazard, to the design of buildings and qualification of the equipment. It is understood that these task are complex, and our aim is only to map the needs of expertise required for preparation of the NPP design process. SESA also has a strong educational element in order to train professionals for undertaking design and review tasks. In Finland seismic design is not required in conventional building projects. Therefore the seismic engineering community is small and fragmented in several organizations. One goal of SESA is to bring together the expertise from different organization working with issues related to seismic engineering.

Specific goals in 2012

The original goals of the SESA projects were set before the very serious accident that occurred at the Fukushima Daiichi nuclear power plant following the major earthquake and devastating tsunami of 11 March 2011. At the new situation the quality and the volume of the original mission related to the seismic risk assessment for 2012 - 2014 needed to be reassessed. This included more comprehensive investigation of methods and regional characteristics in Finland.

One goal for 2012 was to determine attenuation functions for the Fennoscandian Shield of peak ground amplitude and amplitudes of different frequencies, based on locally available data. Further, selected software programs for hazard calculations were to be investigated more carefully, and they were planted to be tested with the data available in Finland.
Figure 2.7.4.1. Location of the 145 swarm events analyzed for the study (a) and peak ground displacement (PDG), velocity (PGV) and acceleration (PGA) of the ML2.6 near the source for vertical (Z) and horizontal (N/E) components at different azimuthal directions (degrees) (b)

On the side of the modelling of building structures, the goal for 2012 was to start the investigation of beyond design basis level loads, when considering reserves due to nonlinearity and energy dissipation. For equipment qualification 2012 was the start-up year. Seismic qualification is more fragmented field then design of structures. The available methods, especially for the qualification of installed equipment require the use of large sets of criteria and databases – hence allowing for much larger margin of interpretation. The aim of
the work in 2012 was to summarize available component qualification databases and qualification procedures available in codes of practice (IEEE 344\(^2\), RCC-E\(^3\)).

Overarching all subprojects was the goal to organize a public course on the topic of earthquakes, hazard assessment, building design and equipment qualification in Aalto University. The course was planned with 12 lectures, most accompanies by exercises and was directed both to post graduate students and practitioners in the engineering design fields.

**Deliverables in 2012**

- The Fennoscandia earthquake databank at the moment contains 32782 three-component recordings from 1381 seismic events, with sampling rate > 80Hz. This database was used to formulate the first attenuation relationship for small magnitude earthquakes in Finland. **Deliverable 1.2** is reporting on the waveforms collected in the data bank and the developed attenuation function. The regional seismic events are usually characterized using local magnitude ML whereas attenuation relations are given as a function of the moment magnitude Mw. Using the events of the data bank, it has been possible to determine the relation between ML and Mw for regional events in Fennoscandia.

- Given the development in direction of proposing attenuation relationships based on Finnish data sets, one important condition for the software platform was to flexibly incorporate user defined attenuation functions. Several software programs (CRISIS 2007, FRISK88M, OpenSHA, EqHaz, OpenQuake and EZ-FRISK) were tested. Based on the review and testing reported in **Deliverable 1.1**, it was decided that EZ-FRISK will be used as reference software.

- The results of the Finite Element Modelling (FEM) of a complex model of a generic reactor building is reported in **Deliverable 2.1/2.2**. The loading was according to the YVL 2.6\(^4\) guide and a broader frequency content spectra having a sPSA plateau between 4Hz and 40Hz. The aim is to explore the bounds of the expected response, in terms of floor spectra, by using the two loading scenarios. The modelling is conducted at higher level of sophistication than expected in current design in order to answer research questions concerning expected behaviour beyond design basis loads.

- **Deliverable 3.1** is a review of the codes of practice guiding qualification.

- The 12 public lectures delivered on earthquakes, seismic hazard assessment and seismic design of SSC’s in Aalto University forms **Deliverables 1.3, 2.4 and 3.2**. Teaching tasks were shared between personnel from Aalto University, the University of Helsinki and VTT. With 26 hours lectures and 80 hours independent work (4 credit points), the course had about 60 participants. Little more than half were practicing engineers from all areas of the Finnish industry, working with seismic issues in their practice. The course material generated by this exercise is a valuable asset supporting dissemination in the future. The teaching material has been reviewed by the Ad-Hoc group of SESA. However, the slides themselves are not part of the deliverables of SESA (due to the autonomy of the university).

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2 IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations.
4 YVL 2.6 - Maanjaristysten Huomioon Ottaminen Ydinvoimalaitoksissa. STUK, 2001
2.8 Probabilistic risk analysis (PRA)

In 2012 the research area "Probabilistic Risk Analysis (PRA)" consisted of four projects: Extreme weather and nuclear power plants (EXWE), Risk assessment of large fire loads (LARGO), PRA development and application (PRADA) and FinPSA knowledge transfer (FINPSA-TRANSFER).

2.8.1 Extreme weather and nuclear power plants (EXWE)

The overall objective of the research in EXWE is to study the frequency, intensity, spatial and temporal variation and the impacts of the extreme weather and sea level events that are relevant for the safety of nuclear power plants. The objectives in 2012 were to examine the occurrence of meteotsunamis and the effect of climate change and wind conditions on the sea levels on the Finnish coast. The aim was also to assess what are the theoretically possible extreme weather events in Finland.

Specific goals in 2012

Specific goals in 2012 were to i) improve the calculations related to the sea level variations and sea level rise due to climate change, ii) investigate meteotsunamis and explore the frequency of these in the Baltic Sea region, iii) estimate the occurrence of theoretically possible weather extremes such as ice storms and blizzards, extreme heat waves and cold spells by employing the 1200-year long millennium simulations, reanalysed weather datasets and observations, and iv) explore the storm track changes in Europe and estimate the possibility of the extreme winter storms such as “Gudrun” in 8.-9.1.2005 to occur in Finland.

Deliverables in 2012

- Wind-induced changes in sea level variability in the Baltic Sea. The main findings were that the zonal geostrophic wind over the southern Baltic Sea explains 82–88% of the interannual sea level variability, and 76–81% of the intra-annual month-to-month variability on the Finnish coast (Johansson, 2013). The annual sea level maxima on the Finnish coast have increased by 20–30 cm from the 1930s. The increase is partially explained by changes in monthly mean wind conditions, and the changes in wind conditions also explain changes in seasonal sea level variability. High sea levels that are exceeded a few weeks per year or less have increased, especially in winter (January–March). An increase in the zonal wind contributed an increasing trend of 0.5–1.2 mm/yr in the sea levels during the 20th century. The accelerating mean sea level trend since 1990s was however not related to regional wind conditions. Based on the climate change projections, the average changes in wind conditions will result in 6–7 cm higher sea levels, the full scenario range extending from a 4 cm decline to a 19 cm increase by the end of this century.

- A review on meteotsunamis and a report on the occurrence of meteotsunamis in the Baltic Sea. Meteotsunamis are long waves in the tsunami frequency band caused by mesoscale atmospheric disturbances, moving above the sea at a resonant speed (Monserrat et al. 2006). In addition, shoaling, focusing by refraction as well as harbor resonance effects are required to amplify the small open sea waves into sizeable waves at the shoreline. Meteotsunami waves occur more or less regularly in some areas of the world ocean, where conditions favor their formation, and under certain conditions the waves can reach destructive dimensions. The meteotsunami phenomenon is known on the Swedish and Finnish coasts under the name sjösprång, and strongest known meteotsunamis have been
1–1.5 m high (Renqvist 1926). In addition to the literature review, three recent meteotsunami cases were investigated. Based on the eyewitness reports the meteotsunamis caused rapid sea level fluctuations of up to 1 m and strong currents with varying direction. Sea level measurements at the Finnish tide gauges confirm these observations (Fig. 2.8.1.1), although with considerably smaller amplitude, as the strength of a meteotsunami strongly depends on coastal bathymetry. The meteorological origin of these events was confirmed by using radar data and coastal observations.

Figure 2.8.1.1. Sea level observations with a 1-minute resolution at the Bothnian Bay tide gauges on 8 August 2010 (Pellikka et al. 2013).

- Report on analysis of the success of earlier mean sea level scenarios. Several scenarios of the global sea level rise have been published during the last 30 years. The first scenarios with starting point at 1980 or 1985 have significant differences already at 2012. In this report these early scenarios were compared with the observed global sea level rise. Four estimates of the observed sea level rise were used: two estimates based on satellite data and two based on tide gauge data. While these estimates have some differences they agree more than an order of magnitude better that the scenarios. Most scenarios (about 70%) are higher than the observed sea level and some extreme scenarios predict six times higher growth for the year 2012 than is now observed. All upper limit scenarios are above the observations. All lower limit scenarios are below or agree with the observed sea level rise. Half of the mid scenarios are above and half agree with the observations. We conclude that observed sea level rise shows no signs to be faster than projected by 1990 and rather seems slower.

- Report on the occurrence of weather extremes that have not been observed in Finland. There are no known cases of severe ice storms or heavy “lake-effect” snowstorms in Finland even though freezing drizzle/rain and abundant snow fall has been observed in the past. The eight more significant freezing rain episodes since 1950’s had mostly caused 4-7 mm of freezing rain in Finland which is very little compared to the real ice storms in Canada (30-100 mm). The climate of the past 1200 years was analyzed for detecting the critical extreme weather episodes by using the fully coupled Max Planck Institute Earth System Model (MPI-ESM)
climate model ("Millennium"). The Millennium-dataset indicated that conditions potentially resulting in over 20 mm of freezing precipitation in a day can occur in the southern and western Finland. The potential freezing precipitation conditions lasted up to 66 hours (~almost three days) in the simulations. Most of the significant simulated events occurred in the autumn and spring months. There were no clear signals in the simulations that severe freezing precipitation events would have become more frequent in southern Finland due to climate change. The “lake-effect” snow is a phenomenon where intense snow showers are formed over relatively warm waters that then get advected to the coast. The worst case scenario in Finland could bring approximately 80-110 kg/m² of snow in a few days (similar to Sweden). The most ideal conditions for heavy lake-effect snows exist in November and December. Additionally, the results indicate that the inclusion of Millennium-data significantly narrows the confidence limits of the return period estimates. The return levels of extremely low and high temperatures (Jokinen et al. 2013) that were received indicated that the best estimates of the 1000 year return level extreme temperatures for instance in Lovisa are +34.0 °C and -38 °C whereas the corresponding values based on observations were +32.1 °C and -40.9 °C, respectively.

- A scientific paper about storm classification and return periods of worst European storms in 1960-2010 by using weather data and forest damage archives. Data of actual wind speed measurements and storm damage is not available of all storms but the most severe ones are rather well described in the scientific literature. We aimed at combining the available meteorological data and the known impacts along the storm tracks and then categorizing the storms (Gregow et al. 2013). We limited the analyses to the last 50 years and the available forest damage reports (EFIATLANTIC) and reanalyzed meteorological datasets ERA-40 (Uppala et al. 2005) and ERA-Interim (Dee et al. 2011). With this data a comparison between the worst Finnish and European storms was made. Based on the reanalyzed dataset and the wind speed calculations, the wind gusts in the worst Finnish storms ranged from 33 ms⁻¹ to 36 ms⁻¹ whereas the significant storms in the rest of Europe typically had wind gusts stronger than 37 ms⁻¹ in large areas. Comparison between the European and Finnish storms showed that on the scale from one to five (minor, moderate, high, very high and extreme impact storm) the strong Finnish storms belong to category 1 (minor impact). Only the storm “Mauri” may belong to category two (moderate impact). The most intensive storms on land in Europe were storms “Lothar” 26.12.1999, “Martin” 27.12.1999, “Gudrun” 8.1.2005. The most extreme storms such as storm “Anatol” 3.12.1999 tend to appear over the Northern Atlantic and in these storms the wind gusts can vary between 50-60 ms⁻¹. The return periods for the storms are difficult to assess, but for instance the storm “Anatol” was estimated to occur once in 140 years near Denmark. The storms “Lothar” and “Martin” were estimated to occur once in 90-100 years in Western Europe. The very positive phase in the Northern Atlantic Oscillation (NAO) seems to prevail when the most extreme storms develop over the Northern Atlantic and move quickly over Western and Central Europe towards east. The intensive storms that move from southwest to northeast may form also when the NAO-index is negative.

- Report on the occurrence of storms based on 1000-year climate simulations. Several international studies about the effects of climate change on storm tracks and storm intensities have been made with many different global and regional climate models as well as reanalyzed meteorological datasets the longest ones dating back to 1870’s. The main conclusion is still that storminess has increased and is still expected to increase in Northern Europe. Yet the observed Finnish cool season storms are much weaker than those in Western- and Central Europe. Therefore it is important to investigate whether such storms as “Lothar” (26.12.1999), “Martin” (27.12.1999), “Gudrun” (8.1.2005) could occur in Finland. The return period calculations of the mean winds showed that a once-in-a-
millennium event in the middle of Gulf of Finland is approximately 32-34 ms$^{-1}$ based on the Millennium-data similar to what was observed during the storm “Gudrun”. However, this result does not take into account the small, but very intense low pressure systems (such as storm “Mauri” or smaller) that the model cannot simulate due to its coarse resolution.

References:


2.8.2 Risk assessment of large fire loads (LARGO)

The main objective of the project is the assessment of risks associated with large industrial fire loads within NPPs, such as oil or electrical cables. The results of the project can be used to ensure the fulfillment of defence-in-depth principle in fire protection. Specific requirements set by the amount and nature of the fire, for the elements of the defence-in-depth principle will be studied, including the possibilities for detection and suppression and fire mitigation by structural partitioning. In the project, fire simulation methods are developed towards a validated capability to predict fire size under ventilation controlled conditions and suppression. The ability to evaluate the efficiency of counter measures by plant personnel is examined by further development and practical application of the new fire-HRA method. The sensitivity of digital automation to smoke and heat will be studied to enable the prediction of damages in components and systems for which empirical data is not yet available. The objective of the fire-HRA and device response studies is the estimation of probability of successful safe shutdown.

Specific goals in 2012

Specific goals of the defence-in-depth (DID) assessment were (i) to prepare an application example together with the power companies, and (ii) to finalize the software tool for the
interoperability of CFD and FEM simulation programs in fire-structure interaction. The feasibility of the proposed DID assessment method was investigated by carrying out extended simulations of the previously studied TVO cable room scenario. A need for new types of design curves was identified as a result of long-lasting fires (traveling fires), typical for large spaces with a plenty of fire load. In the finalization of CFD-FEM interoperability tool, called FDS2FEM, the most of the work was done within FIRE-RESIST EU-project (not related to nuclear field) but the application in this context is also foreseen (Figure below, left).

Concerning the response of the digital automation hardware on smoke and heat, the 2012 goal was to design a test setup and make a plan for specific tests to be carried out in 2013. This work did not proceed for two reasons:

1) Decisions on the experimental conditions require knowledge on the digital automation hardware specifications, such as the line-width. The utilities (TVO and Fortum) were not able to specify during 2012 which kind of hardware should be studied.

2) An experimental campaign with identical goal was carried out by NIST, USA in 2011 (sponsored by U.S.NRC). Examining their test apparatus revealed that a successful test campaign would not be feasible with the resources available in LARGO project. Instead, steps to launch a VTT-NRC co-operation on this topic were taken. Actual continuation of the NIST test series is not known.

While the experimental work on the digital-automation hardware did not proceed, VTT decided to focus on the implementation of a soot-induced surface-resistance degradation model into the fire CFD code (FDS). Resources were also transferred to the experimental characterization of the PVC cable used as a fire source in the OECD PRISME experiments.

The third topic of the project was the prediction of heat release rates from large fire sources, such as pools of combustible liquid of cables. This task included the development and maintenance of the FDS program as part of the international collaboration between VTT and NIST. The most significant new simulation capabilities are

- Simulation of swelling materials (observed in some cable materials). Verification tests were added to the FDS verification suite and validation was carried out using public results on intumescent paints.

- A new model for the prediction of liquid pool evaporation rate was finalized and validated using the NFPA data on various liquids (see figure below, right) and OECD PRISME tests burning Hydrogenated TetraPropylene.

Based on the PRISME simulations, the fire model uncertainty statistics were collected (Table below), to be used in the calculation of the simulation outcome uncertainty (NUREG-1934). Both liquid and cable simulations will continue in 2013.
Figure 2.8.2.1. Transferring fire conditions (CFD) to structural analysis (FEM) on the left, liquid pool burning rate validation (right).

Table 2.8.2.1. Fire simulation uncertainty summary in the scenarios similar to OECD PRISME SOURCE and DOOR scenarios.

<table>
<thead>
<tr>
<th>Quantity</th>
<th>Prescribed HRR</th>
<th>Predicted HRR</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Bias $2\sigma_M$</td>
<td>Bias $2\sigma_M$</td>
</tr>
<tr>
<td>Gas temperature</td>
<td>1.29 0.58</td>
<td>1.15 0.50</td>
</tr>
<tr>
<td>Gas concentrations</td>
<td>0.97 0.22</td>
<td>0.93 0.30</td>
</tr>
<tr>
<td>Wall heat flux</td>
<td>1.37 0.77</td>
<td>1.11 0.76</td>
</tr>
<tr>
<td>Wall temperature</td>
<td>1.12 0.26</td>
<td>1.22 0.34</td>
</tr>
</tbody>
</table>

Deliverables in 2012

- Requirement for DID test loads due to the “travelling fires”.
- FDS2FEM software
- Liquid pool simulation uncertainties.
- Enhanced methodology for the simultaneous application of thermogravimetric (TGA) and micro-scale combustion calorimetry (MCC) data in cable fire simulations.
- New pyrolysis model capability for swelling materials in FDS program.
- Inclusion of Finnish test material in OECD PRISME2 cable fire tests (support tests were carried out in 2012 and the actual test will be carried out in 2013.) Characterization of the cable at VTT.
2.8.3 PRA development and application (PRADA)

The objective of the project is to develop methods that are applicable in the risk analysis of nuclear power plants, and to increase the applicability of nuclear power plant risk analyses. The PRADA project consists of two types of subtasks: reliability analysis method development, and issues in level 2 and 3 PRA. Method development occurs in human reliability analysis, reliability analysis of passive systems, and the application of imprecise probability concepts to PRA.

Specific goals in 2012

Specific goals in 2012 include participation in the EXAM-HRA co-operation project Phase 2, where human reliability analysis practices are compared among German, Swedish, Finnish and now also Swiss nuclear power plants. Another goal for HRA research was to review the HRA methods for digitalized control rooms.

The goal in dynamic PRA subtask was to conduct method development in co-operation with KTH (Kungliga Tekniska Högskolan) and Scandpower of Sweden. This included also the continuation of the international “Integrated Deterministic and Probabilistic Safety Analysis” IDPSA Network based on the DPSA Workshop organised by PRADA in Fall 2011.

In level 3 PRA, the aim is to enhance the capabilities of probabilistic severe accident consequence analysis. In 2012, the main objective was to achieve a common Nordic approach to limited level 3 assessment methodology. The target is to investigate and develop the calculation approach for risk evaluation in the case of a radioactive release to the environment. Methods to estimate risk to the society in addition to an individual are studied taking into account realistic approach. The goal is to be able to calculate doses for an individual representing average living conditions. A common research plan was to be submitted to NKS with Swedish partners Risk Pilot, Scandpower and ES-konsult, who will have a parallel project in Sweden funded by Nordic PSA Group’s Swedish partners.

Another goal on level 3 was to assess interface requirements between levels 2 and 3, and ponder how level 3 computations should best be carried out.

In subtask of passive systems reliability analysis, the bulk of the work was performed in conjunction with a master’s thesis. The safety system of interest was chosen to be Passive Autocatalytic Recombiners (PAR) which are designed for hydrogen management in severe accident conditions. The work introduced a framework (see Figure 2.8.3.1) of how to incorporate reliability analysis of a safety device into a level 2 PSA study. The case part of the study employed Loviisa NPP which has PARs included in its severe accident management plan. Deterministic MELCOR simulations were performed in order to support reliability modelling. Results from PAR reliability analysis were utilized in the Loviisa containment event tree which was modelled with SPSA code. The overall results of the level 2 PSA model highlighted the importance of efficient hydrogen management strategy.
Figure 2.8.3.1. An illustrative scheme of how to conduct a reliability assessment of a safety system and implement it in a CET analysis.

In the imprecise probabilities subtask, the goal was to develop a framework for considering the impact of uncertainty about reliability parameters in fault tree analysis. This was used for analyzing the impact of uncertainty of the aging parameter of a component (Figure 2.8.3.2), which causes the prioritization of the components' failures to vary in time (Figure 2.8.3.3).

Another goal was to compute bounds on the rankings that the event may have when prioritizing them using a risk importance measure. Approximative bounds have been computed and a framework for computing the exact bounds have been developed (manuscript underway).
The framework has also been used for analysing uncertainties of a subsystem of a nuclear power plant with 232 basic events and 1600 minimal cutsets (Figure 2.8.3.3).

Figure 2.8.3.11: Part of the dominance graph of a large fault tree. The nodes of the graph represent basic events of a fault tree, and the arrows represent dominance relations.

**Deliverables in 2012**

- Seminar of main results of phase 2 of EXAM-HRA held on 21.3.2012 at Arlanda.
- An evaluation guide on operator actions in PSA with regard to plant specific features (Phase 2 report).
- Literature review on challenges of human reliability analysis in digitalization of nuclear power plant control rooms
- Euratom FP7 project proposal Integrated Deterministic and Probabilistic Safety Analysis IDPSA submitted in March 2012, funding decision was negative.
- Project proposal on Deterministic-Probabilistic Safety Analysis Methodology submitted to NKS. It approved the proposal.
- Steam explosion case study plan for the development of IDPSA methodology.
- A research plan (project proposal) on a limited level 3 assessment methodology submitted to NKS. It approved the proposal with the suggested funding.
- A research report on level 3 computations from a software architecture point of view.
• A M.Sc. thesis was written on reliability analysis of passive autocatalytic recombiners. The reliability analysis results were utilized in a containment event tree model constructed for Loviisa NPP, according to a framework introduced in the thesis.

• A conference presentation of the developed framework for imprecise probabilities.

• An article based on the framework has been accepted for publication in a scientific journal. Computations with a larger fault tree have confirmed the soundness of the framework for analysing uncertainties on fault trees with a large number of components and minimal cut sets. These findings are reported in an article that is underway.

2.8.4 FinPSA knowledge transfer (FINPSA-TRANSFER)

FinPSA and its predecessor SPSA are unique software tools developed by the Finnish Centre for Radiation Safety (STUK) for carrying out probabilistic risk analyses (PRA). STUK started the development of SPSA in 1988 and the trial usage of level 1 part started in 1991. Level 2 part of the code was completed in early 1993. After validation and pilot modelling, the TVO power company completed the first level 2 PRA in 1997. The development of FinPSA was started in 2000 and the software was launched to customers in 2005. The current version of FinPSA supports only level 1 analysis.

Figure 2.8.4.1. FinPSA – the comprehensive risk and reliability analysis tool for full scope PRA modeling.

In order to support FinPSA and SPSA software end users in the future, an agreement was made to move maintenance and development of these software codes from STUK to VTT Technical Research Centre of Finland. A project, FinPSA knowledge transfer (2012–2014), was started. Ensuring the support and development of these tools was stated to the main objective of the project. In addition, a more general objective was stated to develop and demonstrate practice and skills in the development of software products critical to safety.
Specific goals in 2012

The project consisted of five subprojects in 2012. The aim of coordinating subproject was to prepare a working plan for the FinPSA knowledge transfer and to coordinate other subprojects as the whole. A requirements subproject aimed to prepare the requirements specification for the software. The purpose of specification was to contain requirements related to identifying deficiencies of the present software and to identifying new features to the software. The objective of the design subproject was to prepare a proper SW design specification to be used in the further development of the tools. The testing subproject aimed at preparing a testing plan for FinPSA to fulfil the requirements of high quality software. The objective of the last subproject was to prepare a quality assurance plan which fulfils the general requirements for the development of safety analysis software.

 Deliverables in 2012

- Demonstration that FinPSA mainly conforms to related international standards and the FinPSA specific requirements can be linked to framework represented in the standards.
- Identification and documentation of deficiencies of the present software. Identification and analysis of new developing ideas and harmonization of them together with the current software design.
- Re-documentation and recovery of level 1 design rationales covering architecture description, database and unit designs.
- Identification of fifty properties of the software for testing purposes and study of their testing. It was concluded that the main focus of the testing should be on generation of minimal cut sets, numerical calculations and performance testing. In addition, it was concluded that the testing could be automated.
- Literature survey of PRA and software quality assurance standards. A moderate set of general requirements could be constructed to both the quality of development process and the quality of the software product.

2.9 Development of research infrastructure

In 2012 there were two projects focused on the development of research infrastructure: Enhancement of Lappeenranta instrumentation of nuclear safety experiments (ELAINE) and Renewal of hot cell infrastructure (REHOT).

2.9.1 Enhancement of Lappeenranta instrumentation of nuclear safety experiments (ELAINE)

The objective of the project is to increase the quality and quantity of the measured data generated by the Nuclear Safety Research Unit at Lappeenranta University of Technology (LUT) in order to meet the requirements imposed by today’s CFD modelling. Increasing need for the validation of CFD codes requires adoption of new measuring devices and systems for producing high quality experimental data. Earlier experimental data from LUT has been successfully used for CFD validation. The value of this data is based on innovative use of
traditional instrumentation. However, current requirements for experiments and measurements require introduction of advanced instruments, such as PIV, WMS etc.

The amount of data to be handled increases drastically, when adopting advanced instruments. Large amount of data has to be transferred, handled and finally stored. Ensuring the compatibility of the data acquisition systems used in different test facilities is essential to avoid maintaining of multiple software and system architectures.

To ensure the availability of the test facilities as well as the functioning of the systems in different experiment platforms the traditional control and measuring infrastructure has to be up to date. Long delivery times or even the discontinuance of important components may cause unexpected delays in research projects. Thus, the essential replacement components have to available on-site.

The last link in the chain of experimental research is storing the results in archives in an efficient and convenient manner as well as distributing the results in a controlled way. The STRESA database, developed in an EU project mainly by JRC Ispra, was used at LUT for this purpose. However, the support of STRESA has been suspended and the software used in the database was not anymore compatible with the operating systems currently in use at LUT. This is why new replacing software with similar functions as in STRESA has been developed.

**Specific goals in 2012**

Specific goals in 2012 included ordering of the WMS electronics and continuation of the commissioning of the PIV system to improve the measuring capabilities at LUT. The main circulation pumps for PWR PACTEL were introduced in 2012 plant to expand the applicability and to ease the use of the facility, especially in the preparatory phase of the tests. The development of data storage and distribution software EDS (Experiment Data Storage) continued in 2012.

The wire mesh sensor electronics was purchased from Teletronic Rossendorf GmbH after applying the terms of public procurement in supply contracts, since the price of the system exceeds the national threshold value. Due to the long delivery time the shipment was scheduled in January 2013. Planned low pressure test facility was not constructed and the report of the WMS system and its use is postponed to 2013.

Unexpected problems in the commissioning of the PIV system postponed the testing; the problems were mostly solved in the end of 2012 and PIV system was successfully applied in EXCOP project. Preliminary testing for challenging steam bubble collapse was performed with fluorescent particles.

Purchase of the PWR PACTEL main circulation pumps and assorted equipment, such as power feeds, drives, safety automation and mounting pedestals was completed in 2012. Preparation for the installation of the pumps and the assorted equipment to the PWR PACTEL facility was also done in 2012. The installation plans have been finished. Power feeds have been installed, and the automation system has been upgraded so that the pump controls can be integrated to the system. Pedestals, piping etc. have been designed, and ordered from an outside contractor. Installation of the pumps is scheduled for 2013 after the delivery of the pumps in the beginning of 2013.

In 2012 EDS was opened for internal and external use and both user and administrator manuals were written. In addition to the new software, the hardware in the form of the WWW server was replaced and equipped with an UPS-power feed.
Figure 2.9.1.1. Home page of the EDS database.

Deliverables in 2012

- Getting the PIV system in efficient use partly failed due to the problems with hardware of the controlling workstation. However, before the end of the year PIV was successfully applied in EXCOP tests.

- WMS electronics were purchased, but long delivery time combined with the procurement regulations postponed the delivery to 2013.

- Data storage and distribution software is open for internal and external use. Two manual have been written and they will be update when the software is finalized.

- WWW server was replaced and equipped with an UPS-power feed. Data storage and distribution system EDS is running now in this server
Main circulation pumps have been ordered with the assorted equipment, delivery is scheduled in early 2013.

2.9.2 Renewal of hot cell infrastructure (REHOT)

A national research capability is a basic requirement of the safe use of nuclear power in Finland, and significantly contributes to making the exploitation of nuclear energy economical and efficient. Many critical issues concerning plant life management for operating nuclear power plants are related to materials. Present plans for concurrent lifetime extension, power upgrading, and construction of new plants that may employ new materials in new conditions will require investigating and solving problems related to components and structural integrity. Degradation related to the ageing of structures and components is an important aspect of power plant safety, and ageing management requires activities related to the utilization, inspection, surveillance, testing, examination, and degradation mitigation of materials. This drives the on-going effort to better understand the effects of the reactor environment and operating conditions on the strength and integrity of components and structures. With particular regard to materials that have become activated or contaminated by irradiation-related processes, the ALARA principle requires the utilization of hot cell facilities for conducting fracture mechanical and mechanical testing, metallography, analytical microscopy, autoclave testing, etc. as well as the associated material handling and specimen fabrication and preparation facilities for those techniques.

VTT has been hosting the national hot cell infrastructure since it was first constructed and equipped in the 1970’s. However, the current hot labs are neither longer technically up to date nor adequate to all present research requirements, in addition to being housed in a building that is being renovated by its new owners. As such, during 2009-2010 VTT carried out the conceptual engineering design (tarvekartoitus) and the draft design phases (hankesuunnitelma) for the construction of a new building, to bring together all the nuclear research areas at VTT, i.e., radiochemistry, nuclear waste, dosimetry, fusion materials, failure analysis as well as mechanical and microstructural characterisation of structural materials. The new facility would house hot laboratories, hot cells, and office space for about 150 persons.

Specific goals in 2012

The primary objective of the REHOT project is to plan and execute the hot cell portion of the infrastructure renewal, including the planning and making of critical equipment investments for the hot cell facility, and training of the technical personnel that will be staffing the facility. In 2012 the project was divided into four closely-related subprojects:

- Subproject 1 is dedicated to the hot cell design and construction project, which defines and guides the technical construction of the hot laboratory portion of the new building. Central to this are the shielding cells themselves, to be constructed of suitable materials and equipped with appropriate manipulators and devices to provide protection against human radiation exposure in line with the ALARA principle. The goal in 2012 was to define the specific technical design details of the Hot Cell-related portion of the new building, and to place a subcontract for the fabrication and construction work that is to be outsourced. In subsequent years this subproject will be dedicated to the hot laboratory construction phases.
• Subproject 2 specifically defines the research equipment to be installed in the Hot Cells, and designs the investment schedule and installation parameters in conjunction with the facility construction. In 2012 the procurement of the training manipulator components was to be completed and the procurements of the mechanical test station (MTS) and electro-discharge machines would be carried out.

• Subproject 3 is dedicated to the education and training of staff working in the hot cells, and includes the collecting and describing of procedures to be employed in conducting specific tests and processes within the cells remotely by utilizing manipulators, jigs, tools and other in-cell devices.

• Subproject 4 is to assure continuous exchange of information between the VTT project team and the engineering design team chosen for the design and construction work of the Centre for Nuclear Safety building itself, as it pertains to those aspects broader than the hot laboratory portion itself.

![Figure 2.9.2.1. Architect’s rendition of the future VTT Centre for Nuclear safety (image by SARC Arkkitehtitoimisto).](image)

**Deliverables in 2012**

**Hot Cell Design**

• Assessment of the existing draft plan (hankesuunnitelma) identified critical areas for further attention, including the capacity for storage of irradiated materials and waste, practical interface between A and B lab areas, and minimization and handling of waste
water. Several schematic cell layout alternatives were made, taking into consideration material flow requirements. Through the L2 draft design process of the Centre for Nuclear Safety (CNS), the division of activities between the basement facilities and first floor facilities was proposed, discussed and concluded. Likewise, the utilization of some of the building structures for radiation protection was assessed and specified. On the main level the principle is that all radioactive materials will be handled within shielding at all times, enabling standard wall construction of the building.

- Example hot cell design solutions were examined. In conjunction with the HOTLAB 2012 conference, visits were made to Atalante (CEA’s largest hot laboratory devoted to fuel cycle), NUCLAB (a joint CEA-AREVA analytical laboratory), and ISAI (large hot cell for reconditioning of spent fuel from Phénix reactor). The Halden Kjeller hot-cells were also visited, and while there, discussions were held at length with an Idaho National Labs person regarding details of their newly-constructed shielded autoclave test facilities. A visit was also made to the HZDR (Dresden) hot-cell facilities built in 90’s for Griefswald trepan testing. They represent a modern alternative to VTT’s current RPV hot-cell activities. SCK-CEN’s Mol facilities for waste handling were visited in conjunction with an ATS tour group.

- The search for an external consultant for assisting in the Conceptual Design of the hot cell facilities was carried out, and a competitive bidding process was conducted. Merrick & Company (USA) was selected as the primary choice, and Robatel (France) as the alternate.

- Information was gathered regarding the different hot-cell fabrication approaches to consider, ranging from total DIY approach, through to turn-key approach, as well as potential suppliers across that spectrum.

- A centrifuge waste water separation system was designed, and the components of the pilot centrifuge waste water separation system were fabricated, assembled and the system’s functionality was tested.

![Figure 2.9.2.2. Schematic of a row of manipulator-equipped shielded hot cells for characterization of irradiated reactor materials. (image by ROBATEL).](image-url)
Procurement of Hot Cell equipment

- Inventory of current equipment and their technical requirements was conducted (weight, foot-print, volume, electrical connections etc.). This was instrumental in defining space allocation and facility system requirements in the CNS L2 design process. The main bay containing the principle testing and characterization cells will be a rectangular, 9 m high bay 15 m wide and 33 m long atop the basement level, which can accommodate the equipment in shielded cells. An additional 15 m wide and 6 m long allocation of 3 m high space adjacent to one end of the high-bay will contain microscopy activities (TEM, SEM and LOM) with appropriate ventilation and isolation from vibration and electromagnetic interference.

- Information gathering for future procurements was carried out on several fronts, including direct discussions with (8760) Fastems Oy regarding alternatives for utilizing FANUC industrial robots for mechanized handling of radioactive materials between storage location and between cells, with Instron on their impact test (Charpy) device options, and visits to expos like “Kunnossapito 12 (teollisuuden kunnossapitoa, alihankintapalveluja sekä automaatioratkaisuja)” and “Mittausteknologian päivä” and the HOTLAB2012 conference. Information on suppliers are collected into a single catalogue, currently containing over 30 different suppliers, mostly in Europe.

- Discussions were held with Risø decommissioning project regarding lead glass windows available from their hot-cells. Drawings have been received and will be incorporated into the Conceptual Design process. An option to buy the windows is retained.

- Optical microscope was purchased and billed for.

- Supplier for pre-fatigue pulsator was identified and budgetary offer was received. In lieu of pre-fatigue pulsator, decision was made to purchase a more versatile MTS instrument. Competitive bidding process was conducted, and an order was placed for a Landmark Servohydraulic Test System with 370.10 Load Frame and FlexTest 60 Dual UART/Digital Transducer Interface, from MTS SYSTEMS NORDEN AB. Delivery is scheduled for April 2013.

- Electrodischarge machining (EDM) received stage 1 approval and two budgetary offers were received (Fanuc Robocut and GF AgieCharmilles CUT 200). Evaluation of EDM candidates showed that compact, integrated devices are the most common available. That design does not facilitate installation into a hot-cell such that external access is enabled for many of the components that are not necessary to shield. Further discussions have been held with Fanuc and Mistsubishi on possibilities to modify their machines for hot-cell use. Key features are now being defined that will enable a new round of bidding for a hot-cell adaptable device. As a consequence of the delay, the investment is expected to conclude in the 2014 REHOT project.

- Initial discussions were held with potential suppliers of electron microscopes, and several preliminary estimates are available. TEM and SEM specifications have been utilized to specify the laboratory building requirements. A visit to the Aalto University Nanocentre to view their installation solutions guided the decision to include room ventilation optimized for instrument stability, vibration-resistant floor, isolation of external pumps and chillers in an adjacent room, and minimization of electromagnetic interference by cable route selection in the new facility.
• A temporary decision was gained to carry out large investments as normal VTT investments, utilizing the Research Facility for covering the depreciation and interest costs, but such that the amount VTT spends on making the investment (procurement cost) will be included towards VTT’s VYR matching funds. The investment finance schedule was updated to reflect that.

VTT Centre for Nuclear Safety (CNS) - VTT Ydinturvallisuustalo (YTT)

• Senaatti-kiinteistöt selected A-Insinöörit Rakennuttaminen Oy for L2 draft design of CNS building. L2 draft design process kicked off on 7.8. VTT “users” were involved in a total of 18 formal design meetings and over 35 discussion meetings over the course of the 4 month design process, for collating the necessary input for facility requirements and design alternatives.

• To facilitate VTT-Aalto cooperation in the future CNS, a VTT-Aalto discussion was held. 10 professors attended, representing Aalto’s schools of Science, Engineering, and Chemistry. A positive reception prevailed on all sides.

• To further cement a coalition of support among the national interests, the FINSAFE application was submitted for the Academy of Finland’s Research Infrastructure Roadmap effort, featuring the CNS as the flag-ship and joined by the Finnish Jules Horowitz Reactor program (P. Kinnunen), Lappeenranta University of Technology experimental facility (R. Kyrki-Rajamäki), and Aalto University Nuclear Network (F. Tuomisto).
3. Financial and statistical information

The planned and realised volumes of the SAFIR2014 programme in 2012 were 10.052 M€ and 9.945 M€ and 67 and 71 person years, respectively. The major funding partners were VYR with 5.552 M€, VTT with 2.675 M€, NKS with 0.230 M€, Aalto University with 0.228 M€, Fortum with 0.178 M€, TVO with 0.076 M€, and other partners with 1.006 M€. The planned and realised funding by the major funding partners are illustrated in Fig. 3.1. The planned and realised costs by cost category are shown in Figure 3.2. The personnel costs are the major share of yearly expenses.

![Planned and realised funding in 2012](image)

*Figure 3.1. Planned and realised financing of the SAFIR2014 programme in 2012.*

![Planned and realised costs in 2012](image)

*Figure 3.2. Planned and realised costs of the SAFIR2014 programme in 2012.*
Figures 3.3-3.6 illustrate the cost and volume distributions by research area. In these figures, following abbreviations have been used for the research areas: Human for Man, Organisation and Society; Automation for Automation and Control Room; Core for Fuel Research and Reactor Analysis; Thermal for Thermal Hydraulics; Severe for Severe Accidents; Materials for Structural Safety of Reactor Circuits; Concrete for Construction Safety; PRA for Probabilistic Risk Analysis (PRA); and Infra for Development of Research Infrastructure.

**Figure 3.3.** Planned and realised costs by research area in 2012.

**Figure 3.4.** Planned and realised volumes by research area in 2012.

The main difference in the distributions of funding and person-years is due to the area Development of Research Infrastructure, where the share of personnel costs is relatively low compared to infrastructure investments.
The programme produced 309 publications in 2012. 140 of them were research institute reports. The number of publications is listed in table 3.1. Joint reports of two projects are included in both projects’ publication lists, but counted only once in total number of publications at programme level. The average number of publications was 4.3 per person-year, and the average number of scientific publications was 0.4 per person-year. The number of publications varies a lot between projects.
Table 3.1. Publications in the SAFIR2014 projects in 2012.

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Altogether 8 higher academic degrees were obtained in the research projects in 2012, one Doctoral degree and seven Master’s degrees (see Table 3.2). The academic degrees are presented in Appendix 3.

**Table 3.2. Academic degrees obtained in the projects in 2012.**

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* one Master’s thesis jointly in EXCOP and ELAINE
4. Programme management

During the administrative period (January 2012 – March 2013) the SAFIR2014 steering group held seven meetings. The reference groups met six times, including the evaluation meetings of project proposals for 2013. During 2012, more than twenty Ad Hoc groups supported projects and improved information exchange and co-operation between research areas in multidisciplinary topics. The steering group can also nominate small groups to work on specific topical issues. In 2012, a small group was set up to update the Operational management handbook of the SAFIR2014 programme.

The information on the research performed in SAFIR2014 is communicated formally via the quarterly progress reports of the projects, the annual report of the programme and SAFIR2014 www-pages. 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 separate reports. Poster presentations on the overall research programme were given in connection of two international conferences.

In addition to conducting the actual research according to the annual plans, SAFIR2014 will function as an efficient conveyor of information to all organisations operating in the nuclear energy sector in Finland and as an open discussion forum for participation in international projects, allocation of resources and planning of new projects.

Figure 4.1 illustrates the organisation and quality management in the SAFIR2014 programme. The lists of persons involved in the steering and reference groups, as well as programme staff and their main duties are presented in Appendix 5.

Figure 4.1. Organisation and quality management on the SAFIR2014 programme.
5. References


Appendix 1

Publications in the projects in 2012
Managing safety culture throughout the lifecycle of nuclear plants (MANSCU):

Scientific publications


Conference papers


Research institute reports


Others


Sustainable and future oriented expertise (SAFEX2014):

Conference papers


Others


Coverage and rationality of the software I&C safety assurance (CORSICA):

Conference papers


Timo Varkoi, Risto Nevalainen: Integrating different assessment approaches to evaluate safety-critical software development in nuclear domain, article published in EuroSPI 2012 Industrial proceedings and in a one-day Functional safety workshop 27.6.2012.

Research institute reports

Risto Nevalainen, Timo Varkoi: FiSMA 2012-1: Nuclear SPICE PAM for pre-qualification process assessment.

Timo Varkoi, Risto Nevalainen: FiSMA 2012-2: Nuclear SPICE assessment process.

Timo Varkoi: FiSMA 2012-3: Framework to evaluate software reliability based on Nuclear SPICE.


Others


Human-automation collaboration in incident and accident situations (HACAS):

Scientific publications


Conference papers


Research institute reports


Others


Safety evaluation and reliability analysis of nuclear automation (SARANA):

Scientific publications


Conference papers

http://www.vtt.fi/inf/julkaisut/muut/2012/10-Th4-3.pdf

http://www.vtt.fi/inf/julkaisut/muut/2012/10-Th4-1.pdf


**Research institute reports**


**Others**


**Safety requirements specification and management in nuclear power plants (SAREMAN):**

**Conference papers**


**Others**


Tommila, T., Laarni, J. & Savioja, P. Concept of operations (ConOps) in the design of nuclear power plant instrumentation & control systems. Working report, version 2, 68 p.


Criticality safety and transport methods in reactor analysis (CRISTAL):

Scientific publications


Conference papers


Research institute reports


Others


Three-dimensional reactor analyses (KOURA):

Conference papers


Research institute reports


Others


Räty H. Travel report OECD NEA OS-2 benchmark workshop

Räty H. Travel report AER scientific council

Syrjälähti E. Travel report Reactordynsafe workshop

Syrjälähti E. Travel report OECD/NEA WPRS meeting

Development of a Finnish Monte Carlo reactor physics code (KÄÄRME):

Scientific publications


Conference papers


Research institute reports

Others


Neutronics, nuclear fuel and burnup (NEPAL):

Scientific publications


Conference papers


Research institute reports


Others


Extensive fuel modelling (PALAMA):

Conference papers


Viitanen, T. and Tulkki, V. COMBINING REACTOR PHYSICS AND FUEL PERFORMANCE CALCULATIONS, TopFuel2012, Manchester, United Kingdoms, Sept. 3-6 2012.

Research institute reports


Tulkki V. Analysis of cladding creep and lift-off behaviour, VTT Research Report VTT-R-08519-12.

Others


Radionuclide source term analysis (RASTA):

Research institute reports


Others

Enhancement of safety evaluation tools (ESA):

Research institute reports

Inkinen P., ROSA-2 Test 2 and Test 7 Simulations with APROS and TRACE, VTT-R-01925-13

Kolehmainen J., Simulation of the PACTEL non-condensable gas experiments NCG-1, NCG-3, NCg2-04 and NCg2-05 with APROS 5.11.02, VTT-R-07223-12

Kurki J., System-scale Modelling of ROCOM Mixing Test 2.2, VTT-R-06819-12

Silde A., Modelling of passive autocatalytic hydrogen recombinner (AREVA type) in the APROS Containment code, VTT-R-07488-12

Hillberg S., GEKO building condenser experiments with APROS, VTT-R-01195-13.

Others


Experimental studies on containment phenomena (EXCOP):

Scientific publications


Conference papers


Research institute reports


Others


Tanskanen, V., Lauhdutusaltaan suoran kontaktialuhtumisen CFD-mallinnus käyttäen erottuneen virtauksen lauhtumismalleja. ATS Ydintekniikka 2/2012 vol. 41, pp.21-22.


OpenFOAM CFD-solver for nuclear safety related flow simulations (NuFOAM):

Conference papers


Research institute reports


Peltola, J., twoPhaseNuFoam version 0.4, VTT Research Report VTT-R-01098-13, 48 p.+5 app., 2013.


Others


Numerical modeling of condensation pool (NUMPOOL):

Research institute reports


Others


Improvement of PACTEL facility simulation environment (PACSIM):

Research institute reports


Others


PWR PACTEL experiments (PAX):

Research institute reports

Vesa Riikonen, Virpi Kouhia, Otso-Pekka Kauppinen, Loop seal experiments with PWR PACTEL, Research Report, PAX 1/2012, Lappeenranta University of Technology / Nuclear Safety Research Unit, Lappeenranta, 2012, 18 + 7 s.

Others


**Modeling of pressure transients in steam generators (SGEN):**

**Research institute reports**


**Others**


**Uncertainty evaluation for best estimate analyses (UBEA):**

**Research institute reports**


**Others**


**Core debris coolability and environmental consequence analysis (COOLOCE-E):**

**Scientific publications**

Conference papers


Research institute reports


Others


Chemistry of fission products (FISKE):

Research institute reports


Others

Thermal hydraulics of severe accidents (TERMOSAN):

Conference papers

Sevön, T. MELCOR modeling of passive containment cooling system experiment PANDA T1.1. MCAP meeting, Bethesda, Maryland, September 13–14, 2012.

Research institute reports


Others


Transport and chemistry of fission products (TRAFI):

Conference papers


Research institute reports


Others


Reactor vessel failures, vapour explosions and spent fuel pool accidents (VESPA):

Conference papers


Research institute reports


Others


Environmental influence on cracking susceptibility and ageing of nuclear materials (ENVIS):

Research institute reports

Pakarinen, J. ATEM characterization of a failed core basket bolt removed from Loviisa NPP Unit, VTT-R-00647-12, 27.3.2012.

Pakarinen, J. TEM examination of the effect of post-irradiation annealing on 7.7 dpa AISI 304 stainless steel. VTT-R-05228-12, 4.10.2012.

Pakarinen, J. ATEM characterization of a failed core basket bolt removed from Loviisa NPP Unit 1 – part II. VTT-R-05230-12, 6.10.2012.

Pakarinen, J. ATEM characterization of a slice from a 15 dpa AISI 316 stainless steel baffle bolt. VTT-R-05229-12, 8.10.2012.


Others


**Fracture assessment of reactor circuit (FAR):**

**Scientific publications**


**Research institute reports**


Others


Monitoring of the structural integrity of materials and components in reactor circuit (MAKOMON):

Conference papers


Research institute reports

Leskelä, E & Koskinen, A., Ultrasonic tests to compare mechanical and thermal fatigue cracks in 316L plates, VTT Research Report VTT-R-00932-13, 70 p.


Others


RI-ISI analyses and inspection reliability of piping systems (RAIPSYS):

Scientific publications


Conference papers

Cronvall, O., Männistö, I., Kaunisto, K. On applications concerning OECD pipe failure database OPDE. Conference paper 16A-Tu4-4, 11th International Probabilistic Safety Assessment and Management
Conference (PSAM 11) and The Annual European Safety and Reliability Conference (ESREL 2012). Helsinki, Finland, 25–29 June 2012.

Research institute reports


Cronvall, O., Kaunisto, K. Second phase of a study - Effect of initial flaw and load assumptions on risk estimate changes. VTT Research Report VTT-R-08805-12.


Others


Advanced surveillance technique and embrittlement modeling (SURVIVE):

Scientific publications


Conference papers

T.Toyama, A.Kuramoto, Y.Noza, Y.Matsukawa, M.Hasegawa, Y.Nagai, and M.Valo, Effects of post-irradiation annealing and re-irradiation on microstructure in surveillance test specimens of RPV steel studied by 3D-AP and positron annihilation. TMS-2013 symposium, March 4-7, 2013, San Antonio Texas, USA

Research institute reports


Others


Water chemistry and plant operating reliability (WAPA):

Scientific publications


Conference papers


Research institute reports

Saario, Timo, Hydratsiinin syötön vaikutus hiiliteräksen korroosioon. VTT Tutkimusraportti VTT-R-08318-12, December 2012.

Iva Betova, Martin Bojinov, Timo Saario, Stability of colloidal particles and deposit consolidation in nuclear power plants. VTT Research Reports VTT-R-08112-12, November 2012.

Ikäläinen, Tiina; Lehtikuusi, Taru; Peltonen, Seppo; Saario, Timo and Väisänen, Saija. Effect of ammonia, morpholine and ethanolamine on zeta potential of magnetite – results in 2012. VTT Research Report VTT-R-00211-13, January 2013.

Others


Fatigue affected by residual stresses, environment and thermal fluctuations (FRESH):

Research institute reports


Others


Impact2014 (IMPACT2014):

Scientific publications


Research institute reports


Others


Aging management of concrete structures in nuclear power plants (MANAGE):

Conference papers


Research institute reports


Others


Structural mechanics analysis of soft and hard impacts (SMASH):

Scientific publications


Research institute reports

Hostikka, S. CFD simulation of the aircraft impact fire around a nuclear power plant. VTT-R-00662-13. 30.1.2013. 56 p


Others

Seismic safety of nuclear power plants – targets for research and education (SESA):

Conference papers

I. Smedberg, M. Uski, T. Tiira, K. Komminaho, and A. Korja, Intraplate earthquake swarm in Kouvola, southeastern Finland, European Geosciences Union General Assembly 2012 (EGU2012), Vienna | Austria | 22 – 27 April (poster presentation)

Research institute reports

V. Jussila, L. Fülöp, Principles and practice of exploiting beyond elastic-response reserves in structures, Subproject 2, VTT-R-01167-13 (28 p.)

L. Fülöp, Codes of practice guiding qualification of components in NPP, Subproject 3, VTT-R-01217-13 (18 p.)

Others

M. Malm, N. Leso, J. Saari, Subproject 1 - Earthquake Hazard Assessment, Progress Report 2012, AF-Consult (28 p.)


Extreme weather and nuclear power plants (EXWE):

Scientific publications


Conference papers


Research institute reports


Others


Risk assessment of large fire loads (LARGO):

Scientific publications

Mangs, J., Hostikka, S. Vertical flame spread on charring materials at different ambient temperatures. Fire and Materials (in press), 2012. DOI: 10.1002/fam.2127


Conference papers


Research institute reports


Others


PRA development and application (PRADA):

Conference papers


Research institute reports


Karanta, I. Level 3 PSA from a software architecture point of view. VTT Research report VTT-R-01071-13, Espoo, 15 p. (Draft)
Others


FinPSA knowledge transfer (FINPSA-TRANSFER):

Conference papers


Research institute reports


Others


**Enhancement of Lappeenranta instrumentation of nuclear safety experiments (ELAINE):**

**Others**


**Renewal of hot cell infrastructure (REHOT):**

**Research institute reports**


Karlsen, W. Hot-cell equipment investment financing. VTT-R-01379-13, 9p.

**Others**

Administration of the research programme (ADMIRE):

Conference papers


Research institute reports

(URL: http://www.vtt.fi/publications/index.jsp)
Appendix 2

Participation in international projects and networks in 2012
Management of safety culture throughout the lifecycle of nuclear plants (MANSCU):
NKS (Nordic Nuclear Safety) project MOREMO (Modelling resilience for maintenance and outage)
NKS project SADE (Safety culture in design and implementation of technological and organisational solutions)
IAEA Safety Culture in Pre-operational Phases, participation in consultancy and technical meetings and drafting the IAEA publication.
IAEA Technical Meeting on Managing the unexpected. From the perspective of the Interaction between Individuals, Technology and Organisations.

Coverage and rationality of the software I&C safety assurance (CORSICA):
CENELEC Technical Committee 45AX, Instrumentation and control of nuclear facilities

Human-automation collaboration in incident and accident situations (HACAS):
OECD/NEA WGHOF (Working Group on Human and Organizational Factors)
NULIFE (Nuclear Plant Life Prediction) Network of Excellence (Euratom FP6)
IAEA: Consultancy in “Development of a guideline for Individual-Technology-Organisation (ITO) analysis methodology”

Safety evaluation and reliability analysis of nuclear automation (SARANA):
NKS project DIGREL (Guidelines for reliability analysis of digital systems in PSA context)
Nordic PSA Group project IEC61508
OECD/NEA WGRISK (Working Group on Risk Assesment)
OECD/NEA WGRISK Task group DIGREL (Development of best practice guidelines on failure modes taxonomy for reliability assessment of digital I&C systems for PSA)

Criticality safety and transport methods in reactor analysis (CRISTAL):
OECD/NEA NSC (Nuclear Science Committee)
OECD/NEA WPNCS (Working Party on Nuclear Criticality Safety)
OECD/NEA EGUAM (Expert Group on Uncertainty Analysis in Modelling)
OECD/NEA EGBUC (Expert Group on Burn-up Credit Criticality Safety)
AER WG E (Atomic Energy Research, working group E: radwaste, spent fuel and decommissioning)
EWGRD (European Working Group on Reactor Dosimetry)

Three-dimensional reactor analyses (KOURA):
OECD/NEA WPRS (Working Party on Scientific Issues of Reactor Systems + expert groups)
OECD/NEA Oskarshamn-2 (O2) BWR Stability Benchmark for Coupled Code Calculations and Uncertainty Analysis in Modelling
Follow-up of OECD/NEA benchmarks UAM (Uncertainty Analysis in Best-Estimate Modelling for Design, Operation and Safety Analysis of LWRs ) and K3 (Kalinin-3 Coupled Code Calculationsand Uncertainty Analysis in Modelling )
Scientific Council of AER (Atomic Energy Research)
AER working group D on VVER safety analysis
International Network of Excellence in Reactor Dynamics and Reactor Safety: participation to scientific workshop

**Development of a Finnish Monte Carlo reactor physics code (KÄÄRME):**
Serpent User Group
OECD/NEA Expert Group on Advanced Monte Carlo Techniques

**Neutronics, nuclear fuel and burnup (NEPAL):**
Serpent User Group

**Extensive fuel modelling (PALAMA):**
OECD Halden Reactor Project
OECD/NEA Working Group on Fuel Safety
Halden Programme Group Fuel&Materials
Follow-up of the OECD/NEA Cabri Water Loop Project
Follow-up of OECD/NEA benchmarks UAM (Uncertainty Analysis in Best-Estimate Modelling for Design, Operation and Safety Analysis of LWRs ).

**Enhancement of safety evaluation tools (ESA):**
OECD/NEA/CSNI Working Group on the Analysis and Management of Accidents (WGAMA)
OECD/NEA Rig of Safety Assessment (ROSA-2)
OECD/NEA HYdrogen Mitigation Experiments for REactor Safety (HYMERES)
OECD/NEA Primary Coolant Loop Test Facility Phase 3 project (PKL-3)
USNRC/CAMP Code Applications and Maintenance Program (CAMP)
NORTHNET (The Nordic Thermal Hydraulics and Nuclear Safety Network) Roadmap 3
Modelling and Experiments of Direct-Contact Condensation in Pool Geometry
FONESYS, network among thermal-hydraulic system code developers, co-ordinated by Pisa University

**Experimental studies on containment phenomena (EXCOP):**
NKS project ENPOOL (Experimental and numerical studies on suppression pool issues)
NORTHNET Roadmap 3
NURISP (NUclear Reactor Integrated Simulation Project) (Euratom FP7 project)

**OpenFOAM CFD-solver for nuclear safety related flow simulations (NUFOAM):**
Royal Institute of Technology (RIT), Stockholm, Sweden
Numerical modeling of condensation pool (NUMPOOL):
NKS project ENPOOL (Experimental and numerical studies on suppression pool issues)

PWR PACTEL experiments (PAX):
OECD/NEA Primary Coolant Loop Test Facility Phase 3 project (PKL-3)

Uncertainty evaluation for best estimate analyses (UBEA):
OECD/NEA PREMIUM (Post BEMUSE Reflood Models Input Uncertainty Methods) benchmark

Core debris coolability and environmental consequence analysis (COOLOCE-E):
NKS project DECOSE (Debris Coolability and Steam Explosion)
SARNET2 Network of Excellence / Ex-Vessel Debris Formation and Coolability (WP5.3) (Euratom FP7)

Chemistry of fission products (FISKE):
OECD/NEA BIP-2 PROJECT (To investigate the Behaviour of Iodine in support of source term evaluation in case of severe accident in a nuclear reactor)

Thermal hydraulics of severe accidents (TERMOSAN):
U.S.NRC CSARP (Co-operative Severe Accident Research Program)
OECD/NEA THAI-2 (Thermal Hydraulics, Aerosols and Iodine)
OECD/NEA SERENA-2 (Steam Explosion Resolution for Nuclear Applications)

Transport and chemistry of fission products (TRAFI):
Phedbus FP Project
  - Scientific analysis working group (SAWG)
  - Bundle interpretation circle (BIC)
  - Circuit and containment interpretation circle (CACIC)
  - Containment chemistry interpretation circle (CCIC)
Source term separate effect test program (IRSN, CEA, EDF)
  - International source term scientific analysis working group (SAWG)
  - International source term chemistry interpretation circle (CHEMIC)
OECD/NEA STEM (Source Term Evaluation and Mitigation)
NKS project AIAS-2 (Ad-/absorption and desorption/revaporisation behaviour of iodine aerosols on containment surface materials), co-operation with Chalmers University of Technology
Paul Scherrer Institut (PSI), Villingen, Switzerland
ARTIST follow-up Programme (AeRosol Trapping In a Steam generaTor)
SARNET2 Network of Excellence / Source Term work package (WP8) (Euratom FP7)
Reactor vessel failures, vapour explosions and spent fuel pool accidents (VESPA):
OECD/NEA SERENA-2 (Steam Explosion Resolution for Nuclear Applications)
SARNET2 WP5 (Corium and Debris Coolability) (Euratom FP7)
NKS/DECOSE (Debris Coolability and Steam Explosion)

Environmental influence on cracking susceptibility and ageing of nuclear materials (ENVIS):
OECE/NEA CODAP (Component Operational Experience, Degradation and Ageing Programme)
OECD Halden Reactor Project
NULIFE Network of Excellence (Euratom FP6)
Perform 60 project (Euratom FP7)
EPRI Alloy 690 expert group
EPRI ARRM project (Advanced Radiation Resistant Materials)
International co-operative group on environmentally assisted cracking, ICG-EAC
IFRAM, the International Forum on Reactor Ageing Management (NRC coordinated network)

Fracture assessment of reactor circuit (FAR):
NULIFE Network of Excellence (Euratom FP6) / CABINET-project (Constraint and Biaxial Loading Effects and their Interaction considering Thermal Transients)

Monitoring of the structural integrity of materials and components in reactor circuit (MAKOMON):
US-NRC PARENT program (Program to assess the reliability of emerging nondestructive techniques)

RI-ISI analyses and inspection reliability of piping systems (RAIPSYS):
NUGENIA Association Technical Area 8 (TA8), ENIQ (European Network for Inspection and Qualification) Task Group Risk (TGR) activities
NUGENIA Association Technical Area 1 (TA1), Safety and Risk of NPPs activities

Advanced surveillance-techniques and embrittlement modelling (SURVIVE):
International Group for Radiation Damage Mechanisms (IGRDM)
Joint Research Agreement on "Mini-CT Master Curve Round-robin Tests" organized by CRIEPI/Japan. Seven international partners, a four year programme.
International Round-Robin for the Qualification of Miniaturized Charpy Specimens for the Indirect verification of Small-Scale Impact Machines. Organized by NIST/USA, seven international partners, a two year programme.
Co-operation VTT-Tohoku University on APT and PA characterisation of irradiated materials.
Co-operation agreement between VTT and CRIEPI/JAPAN on APT characterisation of irradiated materials. EDF, ORNL and RRC-KI also partners in the work. Draft agreement available.
Helmholtz-Zentrum Dresden-Rossendorf and VTT. Agreement on performance of SANS-measurements on VTT materials.

**Water chemistry and plant operating reliability (WAPA):**
Co-operation with NUGENIA (Nuclear Generation II&III Association) in preparation of project "Magnetite deposition in the secondary circuit of LWRs”

**Impact 2014 (IMPACT 2014):**
International research project IMPACT 2 (medium scale tests with deformable and hard missiles to the concrete wall with reinforcement) with 8 international partners

**Aging management of concrete structures in nuclear power plants (MANAGE):**
OECD/NEA WGIAGE (Working Group on Integrity ans Ageing of Components and Structures) subgroup on Integrity and ageing of concrete structures
OECD/NEA/WGIAGE project “Post-tensioning methodologies for containment buildings”

**Structural mechanics analysis of soft and hard impacts (SMASH):**
Co-operation with IMPACT 2 project (8 international partners)
European reference network for critical infrastructure protection, working group on Resistance of Structures to Explosion Effects

**Risk assessment of large fire loads (LARGO):**
OECD PRISME 2 project (a project to further investigate fire propagation by means of experiments and analyses relevant for nuclear power plant applications)
NKS project POOLFIRE (Prediction and validation of pool fire development in enclosures by means of CFD Models for risk assessment of nuclear power plants).

**PRA development and application (PRADA):**
OECD/NEA CSNI Working Group RISK
EXAM-HRA: Evaluation of Existing Applications and Guidance on Methods for HRA – NPSAG (Nordic PSA group) and German and Swiss partners
IDPSA Integrated Deterministic-Probabilistic Safety Analysis international network and NKS project in co-operation with Scandpower, KTH and Vattenfall
Level 3 PRA: Finnish-Swedish NKS project on Addressing Off-site Consequence Criteria Using PSA Level 3 in co-operation with Scandpower, Risk Pilot and ESKonsult

**Renewal of hot-cell infrastructure (REHOT):**
HOTLAB- International network for hot laboratories and remote handling

**Administration of the research programme (ADMIRE):**
Consultative Committee Euratom Fission
Appendix 3

Academic degrees obtained in the projects in 2012
Human-automation collaboration in incident and accident situations (HACAS)

*Master of Science in Technology:*

Paula Laitio, User interface solutions for supporting operators’ automation awareness in nuclear power plant control rooms, Aalto University School of Science and Technology, 2013 (January).

Neutronics, nuclear fuel and burnup (NEPAL)

*Master of Science in Technology:*

Ville Valtavirta, Designing and implementing a temperature solver routine for Serpent, Aalto University School of Science, 2012.

Experimental studies on containment phenomena (EXCOP)

*Doctor of Technology:*


*Master of Science in Technology:*

Lauri Pyy, Utilization of Particle Image Velocimetry in PPOOLEX Condensation Experiments, Lappeenranta University of Technology, 2012.

PWR PACTEL experiments (PAX)

*Master of Science in Technology:*

Environmental influence on cracking susceptibility and ageing on nuclear materials (ENVIS)

Master of Science in Technology:


Water chemistry and plant operating reliability (WAPA)

Master of Science in Technology:


PRA development and application (PRADA)

Master of Science in Technology:

Taneli Silvonen, Reliability analysis for passive autocatalytic hydrogen recombiners of a nuclear power plant, Aalto University School of Science, 2012.

Enhancement of Lappeenranta instrumentation of nuclear safety experiments (ELAINE)

Doctor of Technology:


Master of Science in Technology:

Lauri Pyy, Utilization of particle image velocimetry in PPOOLEX condensation experiments. Lappeenranta University of Technology, 2012.
Appendix 4

International travels in the projects in 2012
International travels in MANSCU project in 2012

Oedewald, P, IEA conference, Recife, Brasil, February 9-17, 2012


Macchi L., Reiman T., DESIGN meeting, Stockholm, Sweden, March 02, 2012

Reiman, T., MANSCU meeting, Stockholm, Sweden, May, 30, 2012


Macchi L., Pietikäinen, E., Reiman T., DESIGN meeting, Strängnäs, Sweden, September, 3-4, 2012

Oedewald, P., Macchi L. MOREMO meeting, Ringhals Varberg, Sweden, September, 6-7, 2012

Reiman, T., MANSCU meeting, Stockholm, Sweden, December, 14, 2012


International travels in SAFEX2014 project in 2012

Pahkin K. The European Forum to discuss Nuclear Technology Issues, Opportunities & Challenges, ENC2012, 9 -12, Manchester, United Kingdom.

International travels in CORSICA project in 2012


Nevalainen, R. CENELEC SC45AX meeting 4-5.12.2012, Bruxelles, Belgium.

Nevalainen, R. CENELEC SC45AX workshop, 25-27.4.2012, Petten, Netherlands

Varkoi, T. SPICE 2012 conference, 29-31.5.2012, Palma de Mallorca, Spain

Ranta, J., Lötjönen, L. The 5th International Workshop on Applications of FPGA in Nuclear Power Plants, Beijing, China.

International travels in HACAS project in 2012

Koskinen, H., Workshop on Functional Modelling, November 6-7, 2012, Copenhagen, Denmark.
Koskinen, H., The 8th ANS International Topical Meeting on NPIC-HMIT, July 22-26, 2012, San Diego, USA.


Norros, L., OECD/NEA WGHOF meeting, April 4-6, Paris, France.


Norros, L., Workshop on Functional Modelling, November 6-7, 2012, Copenhagen, Denmark.

Norros, L., OECD/NEA WGHOF meeting, September 17-19, Paris, France.

Savioja, P., The 8th ANS International Topical Meeting on NPIC-HMIT, July 22-26, 2012, San Diego, USA.


International travels in SARANA project in 2012

Holmberg, J.-E., WGRisk Task group DIGREL working meeting, February 16–17, 2012, Paris

Holmberg, J.-E., Working Group Risk annual meeting, March 7–9, 2012, Paris


Holmberg, J.-E., 8th International Topical Meeting on Nuclear Plant Instrumentation, Control, and Human-Machine Interface Technologies, 22.-26.7.2012, The Westin, San Diego, California, USA

Holmberg, J.-E., WGRisk Task group DIGREL seminar and working meeting, November 6–8, 2012, Stockholm


Tyrväinen, T., WGRisk Task group DIGREL seminar and working meeting, November 6–8, 2012, Stockholm

Kuismin, T., European Joint Conferences on Theory and Practice of Software (ETAPS 2012), 24 March - 1 April 2012, Tallinn, Estonia

International travels in SAREMAN project in 2012


International travels in CRISTAL project in 2012

Pauli Juutilainen, MCNP training course February 27- March 2nd, 2012, Paris, France

Maria Pusa, NEA/UAM benchmark meeting, May 9-11, 2012, Karlsruhe, Germany

Karin Rantamäki, AER working group E, May 14-17, 2012, Řež, Czech Republic


Karin Rantamäki, OECD-NEA Working Party on Nuclear Criticality Safety (WPNCS) and its expert groups, September 18-21st, 2012, Paris, France

Tom Serén, European Working Group on Reactor Dosimetry (EWGRD), October 10-12, 2012, Aix-en-Provence, France

International travels in KOURA project in 2012

Syrjälähti, E., Meetings of OECD/NEA Working Party on Scientific Issues of Reactor Systems (WPRS) and expert groups EGRPANS & EGUAM, February 15-17, Issy-les-Moulineaux, France

Syrjälähti, E., OECD/NEA O2-1 workshop and AER working group D meeting, May 7-10, Karlsruhe, Germany

Räty H., OECD/NEA O2-1 workshop, May 7-8, Karlsruhe, Germany

Takasuo E., ICONE-20 conference, July 30- August 3, Anaheim, USA.

Syrjälähti E., Scientific workshop on reactor dynamics and safety, September 13-14, Dresden, Germany.

Syrjälähti E., 22nd AER symposium, October 1-5, Prague, Czech.

Räty H., 23rd meeting of AER’s Scientific Council, November 26-27, Budapest, Hungary.

International travels in KÄÄRME project in 2012


Leppänen, J. Serpent workshop at University of California, Berkeley, USA, April 30 – May 2, 2012.


**International travels in NEPAL project in 2012**

Isotalo, A., Serpent International Users Group Meeting, September 19-21, 2012, Madrid, Spain

Valtavirta, V., Serpent International Users Group Meeting, September 19-21, 2012, Madrid, Spain (in collaboration with KÄÄRME)


**International travels in PALAMA project in 2012**


A. Arffman, research visit to IRSN, 1.1.-30.6.2012, Cadarache, France.

A. Arffman, V. Tulkki, TopFuel 2012, 2 - 6 September 2012, Manchester, United Kingdom.

V. Tulkki, OECD Benchmark for Uncertainty Analysis in Best-Estimate Modelling (UAM) for Design, Operation and Safety Analysis of LWRs – Sixth Workshop (UAM-6), 9.-11.5.2012, KIT, Karlsruhe, Germany.

V. Tulkki, HPG meeting, 28.5.-1.6.2012, Lyon, France.


**International travels in ESA project in 2012**


Karppinen I., OECD/PKL3 Programme Review Group meeting, June 19-20, 2012, Erlangen, Germany

Kurki J., seminar on the transfer of competence, knowledge and experience gained through CSNI activities in the field of thermal-hydraulics (THICKET) June 25-30, Paris, France


Inkinen P., Joint PKL2-ROSA2 workshop on analytical activities related to OECD/NEA PKL and ROSA projects, October 15-19, Paris, France

Alku T., Seminar and Training on Scaling Uncertainty and 3D Coupled Code Calculations in Nuclear Technology (SUNCOP) November 5-23, Dubrovnik, Croatia

Hillberg S., U.S.NRC Code Application and Maintenance Program (CAMP) fall meeting, November 7-9, 2012, Washington DC, USA

International travels in EXCOP project in 2012


International travels in NuFOAM project in 2012


International travels in NUMPOOL project in 2012


International travels in PAX project in 2012

Heikki Purhonen, Vesa Riikonen, Virpi Kouhia The 1st Meeting of the Programme Review Group and Management Board of the OECD/NEA PKL Phase 3 Project, AREVA NP, Erlangen, Germany, 19th-20th June 2012.


International travels in UBEA project in 2012

International travels in COOLOCE-E project in 2012

Takasuo, E. 20th International Conference on Nuclear Engineering collocated with the ASME 2012 Power Conference. July 30 – August 3, 2012, Anaheim, California, USA. (NOTE: travel cost covered mainly by the KOURA project.)

International travels in FISKE project in 2012

Kekki, T., 2nd meeting of the OECD/BIP2 Project Programme Review Group, June 18-19, 2012, Aix-en Provence, France

International travels in TERMOSAN project in 2012


Sevón, T., CSARP (Cooperative Severe Accident Research Program) meeting & MCAP (MELCOR Code Assessment Program) meeting, September 10–14, 2012, Bethesda, Maryland, USA.


International travels in TRAFI project in 2012


Auvinen, A., Kärkelä, T., ISTP and SARNET2 follow-up meetings, 11.6-12.6.2012, Aix-en-Provence, France.


Auvinen, A., Kärkelä, T. 5th European Review Meeting on Severe Accident Research (ERMSAR-2012), 21.3-23.3.2012, Cologne, Germany.

Auvinen, A., ISTP and SARNET2 follow-up meetings, 22.11-27.11.2012, Paris, France.

Kärkelä, T., SARNET2 follow-up meeting, 26.11-27.11.2012, Paris, France.


Kalilainen, J., PhD Day at Paul Scherrer Institute (PSI), 8.6-15.6.2012, Villigen, Switzerland.
Kalilainen, J., Visit at Paul Scherrer Institute (PSI), 25.1-3.2.2012, Villigen, Switzerland.

**International travels in VESPA project in 2012**
Könönen, N., TExAS-V training course organised in co-operation with KTH, November 25 - December 7, 2012, Stockholm, Sweden
Könönen, N., OECD SERENA-2 Final Seminar, November 4-8, 2012, Aix-en-Provence, France
Könönen, N., SARNET2 5th meeting, October 4-8, 2012, Budapest, Hungary
Könönen, N., OECD SERENA-2 Programme Review Group meeting, April 10-13, 2012, Paris, France
Könönen, N., SARNET2 4th meeting, March 20-23, 2012, Cologne, Germany

**International travels in ENVIS project in 2012**
Ehrnstén, U., Hänninen H. International Co-operative Group on Environmentally Assisted Cracking (ICG-EAC) meeting, May 12 – 18, 2012, Quebec, Canada
Karlsen, W. Enlarged Halden Programme Group Meeting 2012, October 17 - 18, 2012, Halden, Norway

**International travels in FAR project in 2012**
Andersson, T., Cabinet (Constraint and Biaxial Loading Effects and their Interaction Considering Thermal Transients (WPS)) project meeting, February, 15, 2012, Munich, Germany
Wallin, K., ASTM (American Society for Testing and Materials) Technical Committee E8, Fatigue and Fracture; technical committee meeting and executive committee meeting, May, 7-9, 2012, Phoenix, Arizona, USA
Wallin, K., ASTM (American Society for Testing and Materials) Technical Committee E8, Fatigue and Fracture; technical committee meeting and executive committee meeting, November, 12-14, 2012, Atlanta, Georgia, USA
International travels in MAKOMON project 2012

Haapalainen, J., 18th World Conference on Nondestructive Testing, April 16-20, 2012, Durban, South Africa.

Jäppinen, T., 18th World Conference on Nondestructive Testing, April 16-20, 2012, Durban, South Africa.


Koskinen, A., 18th World Conference on Nondestructive Testing, April 16-20, 2012, Durban, South Africa.

Leskelä, E., 9th International Conference on NDE in Relation to Structural Integrity for Nuclear and Pressurised Components, May 2012, Seattle, USA.

Tuhti, A., 9th International Conference on NDE in Relation to Structural Integrity for Nuclear and Pressurised Components, May 2012, Seattle, USA.

International travels in RAIPSYS project in 2011

Männistö, I., participation in ENIQ meeting, 26 April 2012, Schiphol, the Netherlands.

Cronvall, O., participation in ENIQ meetings, 23-24 October 2012, Schiphol, the Netherlands.


International travels in WAPA project in 2012


Saario, T., NULIFE + ECG-COMON meeting, February 1-2, 2012, Brugge, Belgium

International travels in IMPACT 2014 project in 2012

Vepsä, A, Calonius Kim, ATS excursion, November 11-17, 2012. Holland and Belgium

International travels in MANAGE project in 2012

Vesikari, E., OECD-NEA Working Group on Integrity of Components and Structures - Integrity and Ageing of Concrete Structures (WGIAGE), September 18th, 2012, Paris, France.

Vesikari, E., IABMAS 2012 International Association for Bridge Management and Safety, July 8-12, 2012, Stresa, Italy

International travels in EXWE project in 2012


Jylhä, K. AMAP Climate Scenario Workshop, 16-18 October, Seattle, USA


International travels in LARGO project in 2012


International travels in PRADA project in 2012


Holmberg, J.-E., Integrated Deterministic and Probabilistic Safety Assessment (IDPSA) working meeting, and Nordic PSA Group’s (NPSAG) meeting and seminar, October 1–3, 2012, Stockholm, Sweden


Holmberg, J.-E., European TSO Network meeting, November 23, 2012, Prague, Czech republic

International travels in FinPSA knowledge transfer project in 2012

Männistö, I., Mätäsniemi, T. FinPSA end user interview in ENSI national regulatory body, 5-6th June, 2012, Brugg, Switzerland

Tyrväinen, T., Mätäsniemi, T. Open-PSA Workshop, 10-11th December, 2012, EdF R&D, Clamart, Paris, France
International travels in ELAINE project in 2012

Pyy, L., ATS excursion to Belgium and the Netherlands, November 11-17, 2012, Brussels, Belgium; Middelburg&Nordwijk&Alkmaar, the Netherlands

Pyy, L., DaVis PIV Training, November 26-30, 2012, Goettingen, Germany

Telkkä, J., DaVis PIV Training, November 26-30, 2012, Goettingen, Germany

International travels in REHOT project in 2012

Karlsen, W. and Valo, M., HOTLAB 2012 conference and hot-cell tours, September 22-26, 2012, Avignon, France

Karlsen, W., visit to Halden’s Kjeller hot-cell for technical tour, October 19, 2012

Lydman, J., ATS tour group visit to Belgian and Dutch sites of interest to nuclear field, including a visit to SCK-CEN’s Mol waste handling facilities, November 11–17, 2012

Karlsen, W., visit to HZDR (Dresden) hot-cell facilities for technical tour, January 24, 2012

International travels in ADMIRE project in 2012

Simola, K., Consultative Committee Euratom-Fission informal Working Group meeting, 30 March 2012, Brussels, Belgium

Simola, K., 2nd meeting of the Consultative Committee Euratom-Fission 2012-2013, and informal exchange of views on Fission research activities under Horizon2020, 21 June 2012, Brussels, Belgium

Simola, K., 3rd meeting of the Consultative Committee Euratom-Fission 2012-2013, and informal exchange of views on Fission research activities under Horizon2020, 25 October 2012, Brussels, Belgium

Simola, K. NKS Fukushima seminar 8-9 January 2013, Stockholm, Sweden
Appendix 5

The steering group, the reference groups and the scientific staff of the projects in 2012
### Steering Group of SAFIR2014

<table>
<thead>
<tr>
<th>Person</th>
<th>Organisation &amp; Finnish abbreviation</th>
</tr>
</thead>
<tbody>
<tr>
<td>Marja-Leena Järvinen, Chairperson</td>
<td>Radiation and Nuclear Safety Authority (STUK)</td>
</tr>
<tr>
<td>Keijo Valtonen, Vice Chairperson</td>
<td>Radiation and Nuclear Safety Authority (STUK)</td>
</tr>
<tr>
<td>Arto Kotipelto</td>
<td>Finnish Funding Agency for Technology and Innovation (Tekes)</td>
</tr>
<tr>
<td>Eija Karita Puska</td>
<td>Technical Research Centre of Finland (VTT)</td>
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<tr>
<td>Pentti Kauppinen</td>
<td>Technical Research Centre of Finland (VTT)</td>
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<tr>
<td>Liisa Heikinheimo</td>
<td>Teollisuuden Voima Oyj (TVO)</td>
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<td>Olli Hoikkala</td>
<td>Teollisuuden Voima Oyj (TVO)</td>
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<tr>
<td>Sami Hautakangas</td>
<td>Fortum Power and Heat Oy (Fortum)</td>
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<tr>
<td>Elizaveta Vainonen-Ahlgren</td>
<td>Fortum Power and Heat Oy (Fortum)</td>
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<td>Juhani Hyvärinen</td>
<td>Fennovoima Oy</td>
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<tr>
<td>Rainer Salomaa</td>
<td>Aalto University</td>
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<td>Sanna Syri</td>
<td>Aalto University</td>
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<td>Riitta Kyrki-Rajamäki</td>
<td>Lappeenranta University of Technology (LUT)</td>
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<td>Heikki Purhonen</td>
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<td>Anneli Leppänen</td>
<td>Finnish Institute of Occupational Health (TTL)</td>
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<tr>
<td>Jaana Avolahti</td>
<td>Ministry of Employment and the Economy (TEM)</td>
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<tr>
<td>Lars Skånberg</td>
<td>Swedish Radiation Safety Authority (SSM)</td>
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<tr>
<td>Jorma Aurela, TEM &amp;VYR contact person</td>
<td>Ministry of Employment and the Economy (TEM)</td>
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<td>Harri Heimbürger, Expert</td>
<td>Radiation and Nuclear Safety Authority (STUK)</td>
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<tr>
<td>Kaisa Simola, Director of SAFIR2014, Secretary of the Steering Group</td>
<td>Technical Research Centre of Finland (VTT)</td>
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SAFIR2014 Reference Groups

1. Man, organisation and society

<table>
<thead>
<tr>
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<tr>
<td>Matti Vartiainen, chairperson</td>
<td>Aalto</td>
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<tr>
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<tr>
<td>Tiina Tigerstedt</td>
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<tr>
<td>Mikko Merikari</td>
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<tr>
<td>Anna-Maria Teperi</td>
<td>(HF expert)</td>
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<td>Per-Olof Sandén</td>
<td>SSM</td>
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<tr>
<td>Kirsi Levä, expert member</td>
<td>TUKES</td>
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<td>(3/2012-)</td>
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2. Automation and control room

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### 3. Fuel research and reactor analysis

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<td><strong>Riku Mattila, chairperson</strong></td>
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### 4. Thermal hydraulics

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<td>Asiantuntijajäsen</td>
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5. Severe accidents

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6. Structural safety of reactor circuits

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<td>Gary Marquis (-6/2012)</td>
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<td>Peter Ekström</td>
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7. Construction safety

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<td><strong>Pekka Välikangas, chairperson</strong></td>
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<td>Jari Puttonen</td>
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8. Probabilistic risk analysis (PRA)

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<tr>
<td><strong>Reino Virolainen, chairperson</strong></td>
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### 9. Development of research infrastructure

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<td><strong>Timo Vanttola, chairperson</strong></td>
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<td>Jorma Aurela</td>
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</table>
Project personnel

**Project Management of safety culture throughout the lifecycle of nuclear plants** (MANSCU)
**Turvallisuuskulttuurin hallinta laitosten elinkaaren eri vaiheissa**

Research organisation: VTT
Project manager: Pia Oedewald, VTT

<table>
<thead>
<tr>
<th>Person</th>
<th>Org.</th>
<th>Tasks</th>
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<tbody>
<tr>
<td>Pia Oedewald, MA Psych</td>
<td>VTT</td>
<td>Project manager; leading the theoretical model update; data collection and analysis in MOREMO</td>
</tr>
<tr>
<td>Luigi Macchi, PhD (Safety management)</td>
<td>VTT</td>
<td>Deputy project manager; responsible for subtask 2, DESIGN</td>
</tr>
<tr>
<td>Teemu Reiman, PhD (Psych)</td>
<td>VTT</td>
<td>Researcher tasks in MOREMO and DESIGN</td>
</tr>
<tr>
<td>Paula Savioja M.sc (Eng)</td>
<td>VTT</td>
<td>Researcher tasks in DESIGN, Human factors engineering</td>
</tr>
<tr>
<td>Marja Liinasuo PhD (Psych)</td>
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<td>Researcher tasks in DESIGN, Human factors engineering</td>
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<tr>
<td>Nadezhda Gotcheva PhD (Industrial management)</td>
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<tr>
<td>Elina Pietikäinen, MA Psych</td>
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<tr>
<td>Kaupo Viitanen, Student Psych</td>
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<tr>
<td>Mikael Wahlström, MA Psych</td>
<td>VTT</td>
<td>Researcher tasks in DESIGN</td>
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**Sustainable and future oriented expertise (SAFEX2014)**
**Kestävää ja kehittyvää tulevaisuuden osaamista**

Research organisations: Aalto University and Finnish Institute of Occupational Health (TTL)
Project manager: Eerikki Mäki, Aalto University

<table>
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<tr>
<td>Eerikki Mäki, DrTech</td>
<td>Aalto</td>
<td>Project manager, design of data collecting instruments, quantitative and qualitative data analysis</td>
</tr>
<tr>
<td>Krista Pahkin, LicSocSci</td>
<td>TTL</td>
<td>Deputy project manager, design of data collecting instruments, quantitative and qualitative data analysis</td>
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<tr>
<td>Tiina Kalliomäki-Levanto, DrTech</td>
<td>TTL</td>
<td>Design of data collecting instruments, quantitative and qualitative data analysis</td>
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<tr>
<td>Tanja Kuronen-Mattila, LicTech</td>
<td>Aalto</td>
<td>Design of data collecting instruments, quantitative and qualitative data analysis</td>
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Coverage and rationality of the software I&C safety assurance (CORSICA)
Turvallisuuskriittisten I&C ohjelmistojen täsmällinen ja kattava varmistaminen

Research organisations: VTT and FiSMA ry.

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<td>Evaluations coverage and rationality</td>
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<td>Jukka Ranta, LicTech</td>
<td>VTT</td>
<td>Novel technologies</td>
</tr>
<tr>
<td>Lauri Lötjönen, MSc Student</td>
<td>VTT</td>
<td>Novel technologies (FPGA)</td>
</tr>
<tr>
<td>Jussi Lahtinen, MScTech</td>
<td>VTT</td>
<td>Reading techniques</td>
</tr>
<tr>
<td>Risto Nevalainen, LicTech</td>
<td>FiSMA</td>
<td>Development of Nuclear SPICE process assessment model and assessment process</td>
</tr>
<tr>
<td>Timo Varkoi, LicTech</td>
<td>FiSMA</td>
<td>Development of Nuclear SPICE assessment process Connection between Nuclear SPICE and reliability</td>
</tr>
</tbody>
</table>

Human-automation collaboration in incident and accident situations (HACAS)
Ihmisen ja automaation yhteistyö häiriö- ja hätätilanteissa

Research organisation: VTT
Project manager: Jari Laarni, VTT

<table>
<thead>
<tr>
<th>Person</th>
<th>Org.</th>
<th>Tasks</th>
</tr>
</thead>
<tbody>
<tr>
<td>Jari Laarni, PhD</td>
<td>VTT</td>
<td>Project manager, Operational concept in accident management, Interactive large-screen displays, Automation awareness</td>
</tr>
<tr>
<td>Paula Savioja, MScTech</td>
<td>VTT</td>
<td>Deputy project manager, Procedure usage in accident management, Operational concept in accident management, Safety automation concept</td>
</tr>
<tr>
<td>Leena Norros, Res Prof, PhD</td>
<td>VTT</td>
<td>Procedure usage in accident management, Automation awareness, Participation in WGHOF</td>
</tr>
<tr>
<td>Iina Aaltonen, MScTech</td>
<td>VTT</td>
<td>Procedure usage in accident management</td>
</tr>
<tr>
<td>Hannu Karvonen, MA</td>
<td>VTT</td>
<td>Operational concept in accident management</td>
</tr>
<tr>
<td>Hanna Koskinen, MA</td>
<td>VTT</td>
<td>Interactive large-screen displays</td>
</tr>
<tr>
<td>Jari Lappalainen, LicScTech</td>
<td>VTT</td>
<td>Simulator environment design</td>
</tr>
<tr>
<td>Marja Liinasuo, PhD</td>
<td>VTT</td>
<td>Operational concept in accident management, Automation awareness</td>
</tr>
<tr>
<td>Mikael Wahlström, MA</td>
<td>VTT</td>
<td>Procedure usage in accident management</td>
</tr>
<tr>
<td>Paula Laitio, Diploma worker</td>
<td>VTT</td>
<td>Automation awareness, simulator environment design</td>
</tr>
</tbody>
</table>
**Safety evaluation and reliability analysis of nuclear automation (SARANA)**

*Ydinvoima-automaation turvallisuuden ja luotettavuuden arviointi*

Research organisations: VTT and Aalto University  
Project manager: Janne Valkonen, VTT

<table>
<thead>
<tr>
<th>Person</th>
<th>Org.</th>
<th>Tasks</th>
</tr>
</thead>
<tbody>
<tr>
<td>Janne Valkonen, MScTech</td>
<td>VTT</td>
<td>Project management, Synthesis of different techniques and abstraction levels in nuclear I&amp;C analysis, Formal analysis of large systems</td>
</tr>
<tr>
<td>Jan-Erik Holmberg, DrTech</td>
<td>VTT</td>
<td>Guidelines for reliability analysis of digital systems in PRA context, development of methods for software reliability quantification</td>
</tr>
<tr>
<td>Kim Björkman, MScTech</td>
<td>VTT</td>
<td>Guidelines for reliability analysis of digital systems in PRA context, YADRAT theory and tool development</td>
</tr>
<tr>
<td>Jussi Lahtinen, MScTech</td>
<td>VTT</td>
<td>Formal analysis of large systems, Modular fault models, Synthesis of different techniques and abstraction levels in nuclear I&amp;C analysis</td>
</tr>
<tr>
<td>Tero Tyrväinen, MScTech</td>
<td>VTT</td>
<td>Guidelines for reliability analysis of digital systems in PRA context, YADRAT theory and tool development</td>
</tr>
<tr>
<td>Antti Pakonen, MScTech</td>
<td>VTT</td>
<td>Synthesis of different techniques and abstraction levels in nuclear I&amp;C analysis</td>
</tr>
<tr>
<td>Keijo Heljanko, DrTech</td>
<td>Aalto</td>
<td>System level interfaces and timing issues, Formal analysis of large systems, Model checking tool diversity, Synthesis of different techniques and abstraction levels in nuclear I&amp;C analysis</td>
</tr>
<tr>
<td>Tuomas Launiainen, MScTech</td>
<td>Aalto</td>
<td>System level interfaces and timing issues, Model checking tool diversity, Synthesis of different techniques and abstraction levels in nuclear I&amp;C analysis</td>
</tr>
<tr>
<td>Siert Wieringa MScTech</td>
<td>Aalto</td>
<td>Formal analysis of large systems, Model checking tool diversity</td>
</tr>
<tr>
<td>Lauri Lammi</td>
<td>Aalto</td>
<td>Model checking tool diversity</td>
</tr>
</tbody>
</table>
Safety requirements specification and management in nuclear power plants (SAREMAN)
Turvallisuusvaatimusten määrittely ja hallinta ydinvoimalaitoksilla

Research organisations: VTT and Aalto University
Project manager: Teemu Tommila, VTT, and Tomi Männistö, Aalto University

<table>
<thead>
<tr>
<th>Person</th>
<th>Org.</th>
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<tbody>
<tr>
<td>Antti Pakonen, MScEng</td>
<td>VTT</td>
<td>Use of natural language requirement templates in model checking, concept and conference paper</td>
</tr>
<tr>
<td>Paula Savioja, MScEng</td>
<td>VTT</td>
<td>Systems Usability Case as an example of the general Assurance Case approach</td>
</tr>
<tr>
<td>Teemu Tommila, MScEng</td>
<td>VTT</td>
<td>Conceptual model for safety requirements specification and management, refinements in traceability of design information. The role of Concept of Operations (ConOps) in the design of nuclear power plant instrumentation &amp; control systems, links to the practices of safety demonstrations. A control engineer’s introduction to requirements engineering, summary of research results in a more practical form</td>
</tr>
<tr>
<td>Janne Valkonen, MScEng</td>
<td>VTT</td>
<td>Conceptual models and safety demonstrations, comments to working drafts</td>
</tr>
<tr>
<td>Quang Doan, Master’s Thesis student</td>
<td>Aalto</td>
<td>Analysis of requirements in standards and YVL guides from regulations viewpoint, V-model as a general reference model</td>
</tr>
<tr>
<td>Tomi Männistö, professor, DrTech</td>
<td>Aalto</td>
<td>Specifying requirements in structured natural language</td>
</tr>
<tr>
<td>Mikko Raatikainen, MScTech</td>
<td>Aalto</td>
<td>Specifying requirements in structured natural language. Conceptual model for safety requirements specification and management</td>
</tr>
<tr>
<td>Eero Uusitalo, MScTech</td>
<td>Aalto</td>
<td>Specifying requirements in structured natural language (temporarily with STUK)</td>
</tr>
</tbody>
</table>
Criticality safety and transport methods in reactor analysis (CRISTAL)
Kriittisyysturvallisuus ja kuljetusmenetelmät reaktorianalyysissä

Research organisation: VTT
Project manager: Karin Rantamäki, VTT

<table>
<thead>
<tr>
<th>Person</th>
<th>Org.</th>
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<tbody>
<tr>
<td>Karin Rantamäki, DrTech</td>
<td>VTT</td>
<td>Project management, criticality safety, calculational methods, international co-operation and training courses</td>
</tr>
<tr>
<td>Petri Kotiluoto, PhD</td>
<td>VTT</td>
<td>Calculational methods, international co-operation and training courses</td>
</tr>
<tr>
<td>Maria Pusa, MScTech</td>
<td>VTT</td>
<td>Calculational methods, international co-operation and training courses</td>
</tr>
<tr>
<td>Antti Räty, MSc</td>
<td>VTT</td>
<td>Calculational methods</td>
</tr>
<tr>
<td>Elina Syrjälähti, MScTech</td>
<td>VTT</td>
<td>Calculational methods</td>
</tr>
<tr>
<td>Pauli Juutilainen, MScTech</td>
<td>VTT</td>
<td>Criticality safety</td>
</tr>
<tr>
<td>Tom Serén, LicTech</td>
<td>VTT</td>
<td>Reactor dosimetry, international co-operation and training courses</td>
</tr>
<tr>
<td>Viitanen Tuomas, MScTech</td>
<td>VTT</td>
<td>Criticality safety</td>
</tr>
<tr>
<td>Leveinen Auli, assistant</td>
<td>VTT</td>
<td>Project management (assistant services)</td>
</tr>
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Three-dimensional reactor analyses (KOURA)
Kolmiulotteiset reaktorianalyysit

Research organisation: VTT
Project manager: Elina Syrjälähti, VTT

<table>
<thead>
<tr>
<th>Person</th>
<th>Org.</th>
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<tbody>
<tr>
<td>Syrjälähti Elina, MScTech</td>
<td>VTT</td>
<td>Project manager, BWR modelling, International co-operation</td>
</tr>
<tr>
<td>Hovi Ville, MScTech</td>
<td>VTT</td>
<td>PORFLO development and applications</td>
</tr>
<tr>
<td>Hämäläinen Anitta, DrTech</td>
<td>VTT</td>
<td>BWR modelling, TRAB3D/SMABRE</td>
</tr>
<tr>
<td>Ilvonen Mikko, LicScTech</td>
<td>VTT</td>
<td>PORFLO development and applications</td>
</tr>
<tr>
<td>Manninen Mikko DrTech</td>
<td>VTT</td>
<td>PORFLO applications</td>
</tr>
<tr>
<td>Rintala Jukka, MScTech</td>
<td>VTT</td>
<td>Neutronics</td>
</tr>
<tr>
<td>Räty Hanna, MScTech</td>
<td>VTT</td>
<td>Deputy project manager, BWR modelling, International co-operation</td>
</tr>
<tr>
<td>Taivassalo Veikko, PhLicPhys</td>
<td>VTT</td>
<td>PORFLO applications</td>
</tr>
<tr>
<td>Takasuo Eveliina, MScTech</td>
<td>VTT</td>
<td>PORFLO applications</td>
</tr>
</tbody>
</table>
Development of a Finnish Monte Carlo reactor physics code (KÄÄRME)
Suomalaisen Monte Carlo -reaktorifysiikkakoodin kehittäminen

Research organisation: VTT
Project manager: Jaakko Leppänen, VTT

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<tr>
<th>Person</th>
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<tbody>
<tr>
<td>Jaakko Leppänen, DrTech</td>
<td>VTT</td>
<td>Project manager, main developer of the Serpent code</td>
</tr>
<tr>
<td>Maria Pusa, MScTech</td>
<td>VTT</td>
<td>Code development</td>
</tr>
<tr>
<td>Tuomas Viitanen, MScTech</td>
<td>VTT</td>
<td>Code development</td>
</tr>
<tr>
<td>Ville Valtavirta, MScTech</td>
<td>VTT</td>
<td>Code development</td>
</tr>
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</table>

Neutronics, nuclear fuel and burnup (NEPAL)
Neutroniikka, ydinpolttoaine ja palama

Research organisation: Aalto University
Project manager: Jarmo Ala-Heikkilä, Aalto University

<table>
<thead>
<tr>
<th>Person</th>
<th>Org.</th>
<th>Tasks</th>
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<tbody>
<tr>
<td>Jarmo Ala-Heikkilä, DrTech</td>
<td>Aalto</td>
<td>Project manager</td>
</tr>
<tr>
<td>Seppo Sipilä, DrTech</td>
<td>Aalto</td>
<td>Deputy project manager</td>
</tr>
<tr>
<td>Aarno Isotalo, MScTech</td>
<td>Aalto</td>
<td>Computer simulations, analytical methods</td>
</tr>
<tr>
<td>Risto Vanhanen, MScTech</td>
<td>Aalto</td>
<td>Computer simulations, analytical methods</td>
</tr>
<tr>
<td>Ville Valtavirta, MScTech</td>
<td>Aalto</td>
<td>Computer simulations, analytical methods</td>
</tr>
<tr>
<td>Markus Ovaska, MSc</td>
<td>Aalto</td>
<td>Computer simulations, analytical methods</td>
</tr>
<tr>
<td>Antti Rintala, BScTech</td>
<td>Aalto</td>
<td>Computer simulations, analytical methods</td>
</tr>
<tr>
<td>Pertti Aarnio, DrTech</td>
<td>Aalto</td>
<td>Internal support group</td>
</tr>
<tr>
<td>Rainer Salomaa, Prof.</td>
<td>Aalto</td>
<td>Internal support group</td>
</tr>
<tr>
<td>Mikko Alava, Prof.</td>
<td>Aalto</td>
<td>Internal support group</td>
</tr>
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</table>
### Extensive fuel modelling (PALAMA)
**Polttoaineen laaja-alainen mallinnus**

Research organisation: VTT  
Project manager: Ville Tulkki, VTT

<table>
<thead>
<tr>
<th>Person</th>
<th>Org.</th>
<th>Tasks</th>
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</thead>
<tbody>
<tr>
<td>Anitta Hämäläinen, DrTech</td>
<td>VTT</td>
<td>Validation of the new boundary conditions in FRAPTRAN-GENFLO.</td>
</tr>
<tr>
<td>Asko Arffman, MScTech</td>
<td>VTT</td>
<td>Analysis of reactivity initiated accidents and loss of coolant accidents, implementation and development of SCANAIR code, WGFS RIA benchmark</td>
</tr>
<tr>
<td>Elina Syrjälähti, MScTech</td>
<td>VTT</td>
<td>Investigation of dimensional behaviour of gadolinia-bearing fuel rods.</td>
</tr>
<tr>
<td>Jan-Olof Stengård, MScTech</td>
<td>VTT</td>
<td>Investigation of FRAPTRAN-GENFLO failure criteria.</td>
</tr>
<tr>
<td>Joonas Kättö, BScTech</td>
<td>VTT</td>
<td>Improvement of validation database system for ENIGMA and the code SPACE for using it, creation of new ENIGMA oxidation models.</td>
</tr>
<tr>
<td>Seppo Kelppe, MScTech</td>
<td>VTT</td>
<td>Effects of load follow operation on fuel performance.</td>
</tr>
<tr>
<td>Tuomas Viitanen, MScTech</td>
<td>VTT</td>
<td>Coupling of ENIGMA and Serpent.</td>
</tr>
<tr>
<td>Ville Tulkki, LicTech</td>
<td>VTT</td>
<td>Development of cladding creep models, project manager.</td>
</tr>
<tr>
<td>Timo Ikonen, DrTech</td>
<td>VTT</td>
<td>Uncertainty analysis, integrated rod model development</td>
</tr>
</tbody>
</table>

### Radionuclide source term analysis (RASTA)
**Radionuklidien lähdetermianalyysit**

Research organisation: VTT  
Project manager: Ville Tulkki, VTT

<table>
<thead>
<tr>
<th>Person</th>
<th>Org.</th>
<th>Tasks</th>
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</thead>
<tbody>
<tr>
<td>Anna Nieminen, MScTech</td>
<td>VTT</td>
<td>Accident conditions and consequences</td>
</tr>
<tr>
<td>Antti Räty, MSc</td>
<td>VTT</td>
<td>Reactor physics analysis</td>
</tr>
<tr>
<td>Ville Tulkki, LicTec</td>
<td>VTT</td>
<td>Fuel behaviour analysis</td>
</tr>
<tr>
<td>Seppo Kelppe, MScTech</td>
<td>VTT</td>
<td>Report inspection</td>
</tr>
</tbody>
</table>
**Enhancement of safety evaluation tools (ESA)**
**Turvallisuusanalyysityökalujen kehittäminen**

Research organisation: VTT  
Project manager: Ismo Karppinen, VTT

<table>
<thead>
<tr>
<th>Person</th>
<th>Org.</th>
<th>Task</th>
</tr>
</thead>
<tbody>
<tr>
<td>Ismo Karppinen, MScTech</td>
<td>VTT</td>
<td>Project manager, participation in OECD/GAMA, follow-up of OECD/ROSA2, OECD/PKL2, coordination of Northnet RM3</td>
</tr>
<tr>
<td>Seppo Hillberg, MScTech</td>
<td>VTT</td>
<td>Deputy project manager, validation of APROS and TRACE codes, follow-up of USNRC/CAMP</td>
</tr>
<tr>
<td>Alku Torsti, trainee</td>
<td>VTT</td>
<td>Development of best estimate methodology in APROS</td>
</tr>
<tr>
<td>Risto Huhtanen, MScTech</td>
<td>VTT</td>
<td>Development and validation of CFD calculation methods</td>
</tr>
<tr>
<td>Markku Hänninen, DrTech</td>
<td>VTT</td>
<td>Participation in the FONESYS code developers network</td>
</tr>
<tr>
<td>Pasi Inkinen, MScTech</td>
<td>VTT</td>
<td>Validation of APROS and TRACE codes</td>
</tr>
<tr>
<td>Jarno Kolehmainen, trainee</td>
<td>VTT</td>
<td>Validation of APROS</td>
</tr>
<tr>
<td>Joona Kurki, LicTech</td>
<td>VTT</td>
<td>APROS 3D module testing and validation</td>
</tr>
<tr>
<td>Sampsa Lauerma, trainee</td>
<td>VTT</td>
<td>Validation of APROS</td>
</tr>
<tr>
<td>Jarto Niemi, MScTech</td>
<td>VTT</td>
<td>Development of CFD models</td>
</tr>
<tr>
<td>Ari Silde, MScTech</td>
<td>VTT</td>
<td>Validation of APROS Containment</td>
</tr>
<tr>
<td>Pekka Urhonen, MScTech</td>
<td>VTT</td>
<td>Validation of APROS</td>
</tr>
</tbody>
</table>
**Experimental studies on containment phenomena (EXCOP)**

**Suojarakennuksessa tapahtuvien ilmiöiden kokeellinen tutkimus**

Research organisation: Lappeenranta University of Technology  
Project manager: Markku Puustinen, Lappeenranta University of Technology

<table>
<thead>
<tr>
<th>Person</th>
<th>Org.</th>
<th>Tasks</th>
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<tbody>
<tr>
<td>Markku Puustinen, MScTech</td>
<td>LUT</td>
<td>Project manager, Experiment planning and analysis</td>
</tr>
<tr>
<td>Jani Laine, MScTech</td>
<td>LUT</td>
<td>Deputy project manager, Experiment analysis, Data conversion</td>
</tr>
<tr>
<td>Heikki Purhonen, DrTech</td>
<td>LUT</td>
<td>International tasks, Experiment planning</td>
</tr>
<tr>
<td>Vesa Riikonen, MScTech</td>
<td>LUT</td>
<td>Data acquisition, Experiments</td>
</tr>
<tr>
<td>Antti Räisänen, MScTech</td>
<td>LUT</td>
<td>Instrumentation, Data acquisition, Data conversion, Visualization, Control systems, Experiments</td>
</tr>
<tr>
<td>Vesa Tanskanen, DrTech</td>
<td>LUT</td>
<td>Computer simulations, Experiments</td>
</tr>
<tr>
<td>Harri Partanen, Engineer</td>
<td>LUT</td>
<td>Designing of test facilities, Experiments</td>
</tr>
<tr>
<td>Hannu Pyikkö, Technician</td>
<td>LUT</td>
<td>Construction, operation and maintenance of test facilities, Experiments</td>
</tr>
<tr>
<td>Ilkka Saure, Technician</td>
<td>LUT</td>
<td>Construction, operation and maintenance of test facilities, Experiments</td>
</tr>
<tr>
<td>Lauri Pyy, MScTech</td>
<td>LUT</td>
<td>Assessment of measurement techniques, Experiments</td>
</tr>
<tr>
<td>Joonas Telkkä, MScTech</td>
<td>LUT</td>
<td>Assessment of measurement techniques, Experiments</td>
</tr>
<tr>
<td>Eetu Kotro, Trainee</td>
<td>LUT</td>
<td>Construction, operation and maintenance of test facilities, Instrumentation, Data acquisition, Data conversion, Visualization, Control systems</td>
</tr>
</tbody>
</table>
OpenFOAM CFD-solver for nuclear safety related flow simulations (NuFOAM)
OpenFOAM CFD - ratkaisija ydinturvallisuuden virtaussimulointiin

Research organisations: VTT, Aalto University, Fortum Power and Heat Oy, LUT
Project manager: Timo Pättikangas, VTT

<table>
<thead>
<tr>
<th>Person</th>
<th>Org.</th>
<th>Task</th>
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<tbody>
<tr>
<td>Timo Pättikangas, DrTech</td>
<td>VTT</td>
<td>Project manager, two-phase CFD modeling</td>
</tr>
<tr>
<td>Juho Peltola, MSc</td>
<td>VTT</td>
<td>Two-phase CFD model development and validation</td>
</tr>
<tr>
<td>Jarto Niemi, MSc</td>
<td>VTT</td>
<td>Two-phase CFD model development and validation</td>
</tr>
<tr>
<td>Timo Siikonen, Prof.</td>
<td>Aalto</td>
<td>CFD model development and validation</td>
</tr>
<tr>
<td>Tomas Brockmann, MSc</td>
<td>Aalto</td>
<td>CFD modeling</td>
</tr>
<tr>
<td>Juhaveikko Ala-Juusela, MSc</td>
<td>Aalto</td>
<td>CFD modeling</td>
</tr>
<tr>
<td>Karoliina Ekström, MSc</td>
<td>Fortum</td>
<td>CFD modeling</td>
</tr>
<tr>
<td>Timo Toppila, MSc</td>
<td>Fortum</td>
<td>CFD modeling</td>
</tr>
<tr>
<td>Vesa Tanskanen, DSc</td>
<td>LUT</td>
<td>CFD model development and validation</td>
</tr>
<tr>
<td>Giteshkumar Patel, MSc</td>
<td>LUT</td>
<td>CFD model development and validation</td>
</tr>
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</table>

Numerical modeling of condensation pool (NUMPOOL)
Lauhdutusaltaan numeerinen mallintaminen

Research organisation: VTT
Project manager: Timo Pättikangas, VTT

<table>
<thead>
<tr>
<th>Person</th>
<th>Org.</th>
<th>Task</th>
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<tbody>
<tr>
<td>Timo Pättikangas, DrTech</td>
<td>VTT</td>
<td>Project manager, CFD modeling of condensation pool</td>
</tr>
<tr>
<td>Jarto Niemi, MScTech</td>
<td>VTT</td>
<td>CFD modeling of condensation pool</td>
</tr>
<tr>
<td>Antti Timperi, MScTech</td>
<td>VTT</td>
<td>Modeling of fluid-structure interactions</td>
</tr>
<tr>
<td>Michael Chauhan, MScTech</td>
<td>VTT</td>
<td>FSI calculation of blowdown with a sector model of a BWR</td>
</tr>
</tbody>
</table>

Improvement of PACTEL facility simulation environment (PACSIM)
PACTEL koelaitteiston simulointiylpäröistön kehittäminen

Research organisation: Lappeenranta University of Technology
Project manager: Juhani Vihavainen, Lappeenranta University of Technology

<table>
<thead>
<tr>
<th>Person</th>
<th>Org.</th>
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</tr>
</thead>
<tbody>
<tr>
<td>Juhani Vihavainen, LicTech</td>
<td>LUT</td>
<td>Project manager, PACSIM project, TRACE code modelling and calculations</td>
</tr>
<tr>
<td>Vesa Riikonen, MScTech</td>
<td>LUT</td>
<td>Data management of experiments</td>
</tr>
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</table>
PWR PACTEL experiments (PAX)  
PWR PACTEL kokeet

Research organisation: Lappeenranta University of Technology  
Project manager: Vesa Riikonen, Lappeenranta University of Technology

<table>
<thead>
<tr>
<th>Person</th>
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<th>Tasks</th>
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<tbody>
<tr>
<td>Vesa Riikonen, MScTech</td>
<td>LUT</td>
<td>Project manager, experiment planning, analysis and reporting, data acquisition</td>
</tr>
<tr>
<td>Markku Puustinen, MScTech</td>
<td>LUT</td>
<td>Deputy project manager, experimental work</td>
</tr>
<tr>
<td>Heikki Purhonen, DTech</td>
<td>LUT</td>
<td>International tasks</td>
</tr>
<tr>
<td>Virpi Kouhia, MScTech</td>
<td>LUT</td>
<td>APROS code modeling and calculations, experiment analysis and reporting</td>
</tr>
<tr>
<td>Otso-Pekka Kauppinen, MScTech</td>
<td>LUT</td>
<td>TRACE code modeling and calculations</td>
</tr>
<tr>
<td>Antti Räisänen, MScTech</td>
<td>LUT</td>
<td>Instrumentation, data acquisition, process control, experimental work</td>
</tr>
<tr>
<td>Harri Partanen, Engineer</td>
<td>LUT</td>
<td>Designing of test facilities</td>
</tr>
<tr>
<td>Hannu Pylkkö, Technician</td>
<td>LUT</td>
<td>Construction, operation and maintenance of test facilities, experimental work</td>
</tr>
<tr>
<td>Ilkka Saure, Technician</td>
<td>LUT</td>
<td>Construction, operation and maintenance of test facilities, experimental work</td>
</tr>
<tr>
<td>Eetu Kotro, Trainee</td>
<td>LUT</td>
<td>Construction, operation and maintenance of test facilities, experimental work</td>
</tr>
</tbody>
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Modelling of pressure transients in steam generators (SGEN)  
Ydinvoimalaitosten vaaka- ja pystyhöyrystinten mallintaminen 3D virtauslaskennalla

Research organisations: VTT and Fortum Power and Heat  
Project manager: Timo Pättikangas, VTT

<table>
<thead>
<tr>
<th>Person</th>
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<tbody>
<tr>
<td>Timo Pättikangas, DTech</td>
<td>VTT</td>
<td>Project manager, development of CFD models for steam generators</td>
</tr>
<tr>
<td>Ville Hovi, MScTech</td>
<td>VTT</td>
<td>CFD and APROS modelling of steam generators</td>
</tr>
<tr>
<td>Jarto Niemi, MScTech</td>
<td>VTT</td>
<td>Development of CFD models for steam generators</td>
</tr>
<tr>
<td>Tommi Rämä, MScTech</td>
<td>Fortum</td>
<td>CFD modeling of VVER-440 horizontal steam generator</td>
</tr>
<tr>
<td>Lauri Peltokorpi, MScTech</td>
<td>Fortum</td>
<td>APROS modeling of Loviisa NPP</td>
</tr>
<tr>
<td>Karo Kustonen, MScTech</td>
<td>Fortum</td>
<td>APROS modeling of Loviisa NPP</td>
</tr>
<tr>
<td>Timo Toppila, MScTech</td>
<td>Fortum</td>
<td>CFD modeling of VVER-440 horizontal steam generator</td>
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</table>
Uncertainty evaluation for best estimate analyses (UBEA)
Best-estimate –turvallisuusanalyysiin yhdistetty epävarmuuksien arvioiminen

Research organisation: VTT
Project manager: Joona Kurki, VTT

<table>
<thead>
<tr>
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<tbody>
<tr>
<td>Joona Kurki, Lic.Tech</td>
<td>VTT</td>
<td>Project manager, research on DPSA</td>
</tr>
<tr>
<td>Torsti Alku, Trainee</td>
<td>VTT</td>
<td>BEPU analyses, research on input uncertainties</td>
</tr>
</tbody>
</table>

Core debris coolability and environmental consequence analysis (COOLOCE-E)
Sydänromukeon jäähdyttävyyys ja ympäristövaikutusten arviointi

Research organisation: VTT
Project manager: Eveliina Takasuo, VTT

<table>
<thead>
<tr>
<th>Person</th>
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<tbody>
<tr>
<td>Eveliina Takasuo, MScTech</td>
<td>VTT</td>
<td>Planning and analysis of the COOLOCE experiments, modelling of debris coolability, atmospheric dispersion analysis</td>
</tr>
<tr>
<td>Tuomo Kinnunen, Engineer</td>
<td>VTT</td>
<td>COOLOCE experiments: design, installation and maintenance work</td>
</tr>
<tr>
<td>Taru Lehtikuusi, Technician</td>
<td>VTT</td>
<td>COOLOCE experiments: design, installation and maintenance work</td>
</tr>
<tr>
<td>Stefan Holmström, DrTech</td>
<td>VTT</td>
<td>COOLOCE experiments: design, reporting and management</td>
</tr>
<tr>
<td>Ville Hovi, MScTech</td>
<td>VTT</td>
<td>Modelling of debris coolability and PORFLO code development</td>
</tr>
<tr>
<td>Mikko Ilvonen, LicTech</td>
<td>VTT</td>
<td>Modelling of atmospheric dispersion of radionuclides, Fukushima accident</td>
</tr>
</tbody>
</table>
### Chemistry of fission products (FISKE)

**Fissiotuotteiden kemia**

Research organisation: VTT  
Project manager: Tommi Kekki, VTT

<table>
<thead>
<tr>
<th>Person</th>
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<tbody>
<tr>
<td>Tommi Kekki, MSc</td>
<td>VTT</td>
<td>Project manager, OECD/BIP2 project, nitric acid formation</td>
</tr>
<tr>
<td>Lamminmäki Suvi, MSc</td>
<td>VTT</td>
<td>Nitric acid formation, GEMINI2</td>
</tr>
<tr>
<td>Maija Lipponen, MSc</td>
<td>VTT</td>
<td>Nitric acid formation</td>
</tr>
<tr>
<td>Jaana Rantanen</td>
<td>VTT</td>
<td>Nitric acid formation</td>
</tr>
<tr>
<td>Könönen Niina, MScTech</td>
<td>VTT</td>
<td>MELCOR calculations</td>
</tr>
<tr>
<td>Rossi Jukka, MScTech</td>
<td>VTT</td>
<td>Dose calculations</td>
</tr>
<tr>
<td>Penttilä, Karri, MScTech</td>
<td>VTT</td>
<td>Chempool, Chemsheet, CSFoam viscosity</td>
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<tr>
<td>Köttiluoto Petri, DrTech</td>
<td>VTT</td>
<td>ORIGEN2 source term</td>
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<tr>
<td>Räty Antti, MScTech</td>
<td>VTT</td>
<td>ORIGEN2 source term</td>
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### Thermal hydraulics of severe accidents (TERMOSAN)

**Vakavien onnettomuuksien termohydrauliikka**

Research organisation: VTT  
Project manager: Tuomo Sevón, VTT

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<thead>
<tr>
<th>Person</th>
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<tbody>
<tr>
<td>Tuomo Sevón, MScTech</td>
<td>VTT</td>
<td>Fukushima modelling with MELCOR, CSARP, OECD THAI-2, project management</td>
</tr>
<tr>
<td>Anna Nieminen, MScTech</td>
<td>VTT</td>
<td>PCCS calculations with ASTEC</td>
</tr>
<tr>
<td>Veikko Taivassalo, MScTech</td>
<td>VTT</td>
<td>Fluent calculations of hydrogen combustion experiments in THAI-2</td>
</tr>
<tr>
<td>Ilona Lindholm, MScTech</td>
<td>VTT</td>
<td>OECD SERENA-2</td>
</tr>
<tr>
<td>Niina Könönen, MSc</td>
<td>VTT</td>
<td>OECD SERENA-2</td>
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</table>
Transport and chemistry of fission products (TRAFI)
Fissiotuotteiden kulkeutuminen ja kemia

Research organisation: VTT
Project manager: Teemu Kärkelä, VTT

<table>
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<tr>
<th>Person</th>
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<tbody>
<tr>
<td>Teemu Kärkelä, MScTech</td>
<td>VTT</td>
<td>Iodine experiments, Participation in Phebus, ISTP and STEM projects</td>
</tr>
<tr>
<td>Ari Auvinen, MScTech</td>
<td>VTT</td>
<td>Participation in Phebus, ISTP, ARTIST and STEM projects, Interpretation of results</td>
</tr>
<tr>
<td>Unto Tapper, PhD</td>
<td>VTT</td>
<td>Electron microscopy - iodine experiments</td>
</tr>
<tr>
<td>Raoul Järvinen, Technician</td>
<td>VTT</td>
<td>Construction of experimental facilities</td>
</tr>
<tr>
<td>Suvi Lamminmäki, Research Trainee</td>
<td>VTT</td>
<td>Radiotracer tests - iodine experiments</td>
</tr>
<tr>
<td>Emmi Myllykylä, MSc</td>
<td>VTT</td>
<td>Chemical analysis - iodine experiments</td>
</tr>
<tr>
<td>Jaana Rantanen, Technician</td>
<td>VTT</td>
<td>Chemical analysis - iodine experiments</td>
</tr>
<tr>
<td>Tuula Kajolinna, Engineer</td>
<td>VTT</td>
<td>Gaseous compounds analysis - iodine experiments</td>
</tr>
<tr>
<td>Jarmo Kalilainen, MScTech</td>
<td>VTT</td>
<td>Iodine experiments in primary circuit conditions, Studies on particle deposition in turbulent flow</td>
</tr>
<tr>
<td>Pekka Rantanen, LicTech</td>
<td>VTT</td>
<td>Iodine experiments in primary circuit conditions, Studies on particle deposition in turbulent flow</td>
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Reactor vessel failures, vapour explosions and spent fuel pool accidents (VESPA)
Paineastian puhkeaminen, höyryräjähdysten mallinnus ja käytetyn polttoaineen altaiden onnettomuudet

Research organisation: VTT
Project manager: Niina Könönen, VTT

<table>
<thead>
<tr>
<th>Person</th>
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<tbody>
<tr>
<td>Niina Könönen, MSc</td>
<td>VTT</td>
<td>Project manager, Steam explosion analyses, Spent fuel pool analyses with MELCOR</td>
</tr>
<tr>
<td>Mikko Patalainen, MScTech</td>
<td>VTT</td>
<td>Benchmarking Abaqus against PASULA</td>
</tr>
<tr>
<td>Kari Ikonen, DTech</td>
<td>VTT</td>
<td>Spent fuel pool analyses with PANAMA, Benchmarking Abaqus against PASULA</td>
</tr>
<tr>
<td>Ilona Lindholm, MScTech</td>
<td>VTT</td>
<td>Spent fuel pool analyses</td>
</tr>
</tbody>
</table>
Environmental influence on cracking susceptibility and ageing on nuclear materials (ENVIS)
Ympäristön vaikutus ydinvoimalaitosmateriaalien murtumiseen ja vanhenemiseen

Research organisations: VTT and Aalto University
Project manager: Ulla Ehrnstén, VTT

<table>
<thead>
<tr>
<th>Person</th>
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<tbody>
<tr>
<td>Ulla Ehrnstén, MScTech</td>
<td>VTT</td>
<td>Project manager, characterisation of materials, international co-operation, mentor</td>
</tr>
<tr>
<td>Seppo Tähtinen, MScTech</td>
<td>VTT</td>
<td>Deputy project manager, participating in development of biaxial creep equipment</td>
</tr>
<tr>
<td>Matias Ahonen, MScTech</td>
<td>VTT</td>
<td>Responsible for performing fracture toughness tests</td>
</tr>
<tr>
<td>Pasi Väisänen, Technician</td>
<td>VTT</td>
<td>Performance of super slow rate tests in simulated LWR environment</td>
</tr>
<tr>
<td>Moilanen Pekka, DrTech</td>
<td>VTT</td>
<td>Responsible for biaxial creep equipment design</td>
</tr>
<tr>
<td>Aaltonen Pertti, MScTech</td>
<td>VTT</td>
<td>Responsible for investigations of aged Stellite 6 material</td>
</tr>
<tr>
<td>Wade Karlsen, DrTech</td>
<td>VTT</td>
<td>International co-operation, mentor</td>
</tr>
<tr>
<td>Janne Pakarinen, DrTech</td>
<td>VTT</td>
<td>Responsible for characterisation of irradiated stainless steels</td>
</tr>
<tr>
<td>Pasi Kuivalainen, Research Engineer</td>
<td>VTT</td>
<td>Responsible for autoclave testing and mentoring autoclave workers until resigning from VTT in summer 2012</td>
</tr>
<tr>
<td>Juha-Matti Autio, MScTech</td>
<td>VTT</td>
<td>Characterisation of SSSRT specimens and mock-ups</td>
</tr>
<tr>
<td>Jyrki Ranta, Research Engineer</td>
<td>VTT</td>
<td>Participating in SSSR tests, training in autoclave testing</td>
</tr>
<tr>
<td>Heikki Keinänen, Senior MScTech</td>
<td>VTT</td>
<td>Performing FE-modelling</td>
</tr>
<tr>
<td>Päivi Karjalainen-Roikonen, MScTech</td>
<td>VTT</td>
<td>Performing fracture toughness results analyses</td>
</tr>
<tr>
<td>Risto Ilola, DrTech</td>
<td>Aalto</td>
<td>Aalto project manager</td>
</tr>
<tr>
<td>Mykola Ivanchenko, DrTech</td>
<td>Aalto</td>
<td>Instructor on Master on Science work on fatigue, performance of fatigue tests</td>
</tr>
<tr>
<td>Helka Apajalahti, student</td>
<td>Aalto</td>
<td>Master of Science work on the role of DSA in fatigue</td>
</tr>
<tr>
<td>Mouginot Roman, (student) MSc</td>
<td>Aalto</td>
<td>Master of Science work on dissimilar metal welds</td>
</tr>
<tr>
<td>Tapio Saukkonen, MScTech</td>
<td>Aalto</td>
<td>Responsible for EBSD, FEGSEM and nano-indentation investigations</td>
</tr>
<tr>
<td>Anssi Brederholm, MScTech</td>
<td>Aalto</td>
<td>Responsible for preparation of materials used in tests</td>
</tr>
<tr>
<td>Erno Soinila, MScTech</td>
<td>Aalto</td>
<td>Responsible for investigations on thermal ageing of duplex stainless steels</td>
</tr>
<tr>
<td>Hannu Hänninen, Prof.</td>
<td>Aalto</td>
<td>Scientific support, mentor</td>
</tr>
</tbody>
</table>
Fracture assessment of reactor circuit (FAR)
Reaktoripiirin murtumisriskin arviointi

Research organisation: VTT
Project manager: Päivi Karjalainen-Roikonen, VTT

<table>
<thead>
<tr>
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<tbody>
<tr>
<td>Päivi Karjalainen-Roikonen, MScTech</td>
<td>VTT</td>
<td>Low constraint, DMW, LBB</td>
</tr>
<tr>
<td>Heikki Keinänen, MScTech</td>
<td>VTT</td>
<td>LBB, FEM methods</td>
</tr>
<tr>
<td>Juha Kuutti, MScTech</td>
<td>VTT</td>
<td>XFEM</td>
</tr>
<tr>
<td>Kalle Kaunisto, MScTech</td>
<td>VTT</td>
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<tr>
<td>Tapio Planman, MScTech</td>
<td>VTT</td>
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<tr>
<td>Tom Andersson</td>
<td>VTT</td>
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<tr>
<td>Pekka Nevasmaa, MScTech</td>
<td>VTT</td>
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Monitoring of the structural integrity of materials and components in reactor circuit (MAKOMON)
Materiaalien ja komponenttien eheyden monitorointi

Research organisation: VTT
Project manager: Tarja Jäppinen, VTT

<table>
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<tr>
<td>Tarja Jäppinen, LicTech</td>
<td>VTT</td>
<td>Project manager, Eddy current applications</td>
</tr>
<tr>
<td>Ari Koskinen, MScTech</td>
<td>VTT</td>
<td>Deputy project manager, Ultrasonic reflectors</td>
</tr>
<tr>
<td>Jonne Haapalainen, MSc</td>
<td>VTT</td>
<td>Ultrasonic simulations and reflectors, Digital radiography</td>
</tr>
<tr>
<td>Esa Leskelä, MScTech</td>
<td>VTT</td>
<td>Ultrasonic simulations and reflectors</td>
</tr>
<tr>
<td>Stefan Sandlin, MSc</td>
<td>VTT</td>
<td>New ultrasonics methods</td>
</tr>
<tr>
<td>Kari Lahdenperä, LicScTech)</td>
<td>VTT</td>
<td>Eddy current applications</td>
</tr>
<tr>
<td>Antti Tuhti, Engineer</td>
<td>VTT</td>
<td>Ultrasonic reflectors, Digital radiography</td>
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<tr>
<td>Matti Sarkimo, LicScTech</td>
<td>VTT</td>
<td>Ultrasonic simulations</td>
</tr>
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</table>
RI-ISI analyses and inspection reliability of piping systems (RAIPSYS)
Putkistojärjestelmien RI-ISI analyysit ja tarkastusten luotettavuus

Research organisation: VTT
Project manager: Otso Cronvall, VTT

<table>
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<tbody>
<tr>
<td>Otso Cronvall, LicTech</td>
<td>VTT</td>
<td>Project manager, risk informed in-service inspection methodology, probabilistic fracture mechanics</td>
</tr>
<tr>
<td>Jouni Alhainen, MScTech</td>
<td>VTT</td>
<td>Probabilistic fracture mechanics software development</td>
</tr>
<tr>
<td>Kalle Kaunisto, MScTech</td>
<td>VTT</td>
<td>Database applications, structural reliability software development</td>
</tr>
<tr>
<td>Ilkka Männistö, MScTech</td>
<td>VTT</td>
<td>Markov modelling for piping break risks and POD sensitivity analyses</td>
</tr>
<tr>
<td>Taneli Silvonen, MScTech</td>
<td>VTT</td>
<td>Piping break risk analyses</td>
</tr>
<tr>
<td>Ari Vepsä, MScTech</td>
<td>VTT</td>
<td>Deputy project manager, probabilistic simulations and sampling methods</td>
</tr>
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Advanced surveillance-techniques and embrittlement modelling (SURVIVE)
Kehittynyt surveillance-tekniikka ja materiaaliominai-suuksien luotettava mallintaminen

Research organisation: VTT
Project manager: Matti Valo, VTT

<table>
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<tr>
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<tbody>
<tr>
<td>Matti Valo, MScTech</td>
<td>VTT</td>
<td>Planning, analysis, consultant, reporting</td>
</tr>
<tr>
<td>Petteri Lappalainen, MSc</td>
<td>VTT</td>
<td>Resistivity measurements</td>
</tr>
<tr>
<td>Anssi Laukkanen, MScTech</td>
<td>VTT</td>
<td>Modelling, calculations</td>
</tr>
<tr>
<td>Jari Lydman, engineer</td>
<td>VTT</td>
<td>Mechanical testing, EB-welding, metallography and HV-tests</td>
</tr>
<tr>
<td>Tuomo Lappalainen, engineer</td>
<td>VTT</td>
<td>Mechanical testing</td>
</tr>
</tbody>
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Water chemistry and plant operating reliability (WAPA)
Vesikemia ja laitosten käyttövarmuus

Research organisation: VTT
Project manager: Timo Saario, VTT

<table>
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<tr>
<th>Person</th>
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<tr>
<td>Taru Lehtikuusi, Research Assistant</td>
<td>VTT</td>
<td>Chemistry control</td>
</tr>
<tr>
<td>Seppo Peltonen, Research Assistant</td>
<td>VTT</td>
<td>Equipment design</td>
</tr>
<tr>
<td>Timo Saario, DrTech</td>
<td>VTT</td>
<td>Project manager, planning and reporting</td>
</tr>
<tr>
<td>Mikko Vepsäläinen, MSc</td>
<td>VTT</td>
<td>Deputy project manager, modelling of magnetite deposition</td>
</tr>
<tr>
<td>Saija Väisänen, MSc</td>
<td>VTT</td>
<td>Measurement of zeta-potential at PWR secondary side conditions</td>
</tr>
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Fatigue affected by residual stresses, environment and thermal fluctuations (FRESH)
Jäännösjännitysten, ympäristön ja lämpötilavaihteluiden vaikutukset väsymiseen

Research organisation: VTT
Project manager: Michael Chauhan, VTT

<table>
<thead>
<tr>
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<tbody>
<tr>
<td>Michael Chauhan, MScTech</td>
<td>VTT</td>
<td>Project manager, $F_{en}$ and fatigue computation, literature review</td>
</tr>
<tr>
<td>Heikki Keinänen, MScTech</td>
<td>VTT</td>
<td>Deputy project manager , welding simulation</td>
</tr>
<tr>
<td>Otso Cronvall, LicTech</td>
<td>VTT</td>
<td>Literature review</td>
</tr>
<tr>
<td>Antti Timperi, MScTech</td>
<td>VTT</td>
<td>CFD computation, numerical modelling</td>
</tr>
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</table>
Impact 2014 (IMPACT 2014)

Research organisation: VTT
Project manager: Ilkka Hakola, VTT

<table>
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<tr>
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<tbody>
<tr>
<td>Ilkka Hakola, MScTech</td>
<td>VTT</td>
<td>Project manager, 1 2 3 4 5</td>
</tr>
<tr>
<td>Ari Vepsä, MSc Tech</td>
<td>VTT</td>
<td>Deputy project manager 1 2 3 4 5</td>
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<tr>
<td>Kim Calonius, MScTech</td>
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<tr>
<td>Matti Halonen, MScTech</td>
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<tr>
<td>Jouni Hietalahti, Research eng.</td>
<td>VTT</td>
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<tr>
<td>Erkki Järvinen, MScTech</td>
<td>VTT</td>
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<tr>
<td>Kari Korhonen, Research eng</td>
<td>VTT</td>
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<tr>
<td>Eila Lehmus, MScTech</td>
<td>VTT</td>
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<tr>
<td>Jukka Mäkinen, Research eng</td>
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<td>Mikko Patalainen, MScTech</td>
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<tr>
<td>Arja Saarenheimo, LicScTech</td>
<td>VTT</td>
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<tr>
<td>Erja Schlesier, Assistant</td>
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<tr>
<td>Ville Sjöblom, Research eng</td>
<td>VTT</td>
<td>1 2 4 5</td>
</tr>
<tr>
<td>Secil Uanic, Assistant</td>
<td>VTT</td>
<td>5</td>
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</table>

Tasks: 1 Test apparatus; 2 Measurements, 3 Pre-calculations, 4 Testing, 5 Archiving

The material tests and casting have been done by VTT Expert Services Oy. Pre-stressing of concrete walls have been done by the company of Tensicon consulting.
Aging management of concrete structures in nuclear power plants (MANAGE)
Ydinvoimaloiden betonirakenteiden ikääntymisen hallinta

Research organisations: VTT and Aalto University
Project manager: Erkki Vesikari, VTT (-12/2012) and Miguel Ferreira, VTT (12/2012-)

<table>
<thead>
<tr>
<th>Person</th>
<th>Org.</th>
<th>Tasks</th>
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</thead>
<tbody>
<tr>
<td>Erkki Vesikari, LicTech</td>
<td>VTT</td>
<td>Project manager, Planning of the platform, application and interfacing of the program ServiceMan, Condition analyses based on samples.</td>
</tr>
<tr>
<td>Miguel Ferreira, PhD</td>
<td>VTT</td>
<td>Project manager, Planning of the platform, application and interfacing of the program ServiceMan, condition analyses based on samples.</td>
</tr>
<tr>
<td>Mikko Tuomisto, BSc</td>
<td>VTT</td>
<td>Planning of the MANAGE platform. Design of interfaces between central database and analysis tools.</td>
</tr>
<tr>
<td>Olli Stenlund, MSc</td>
<td>VTT</td>
<td>Planning of the MANAGE platform. Design of interfaces between central database and analysis tools.</td>
</tr>
<tr>
<td>Petr Hradil, PhD</td>
<td>VTT</td>
<td>Visualisation of structures.</td>
</tr>
<tr>
<td>Esko Sistonen, DrTech</td>
<td>Aalto</td>
<td>Condition analyses based on samples.</td>
</tr>
<tr>
<td>Jukka Piironen, MScTech</td>
<td>Aalto</td>
<td>Condition analyses based on samples.</td>
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Structural mechanics analysis of soft and hard impacts (SMASH)
Rakenteiden mekaniikan menetelmät pehmeiden ja kovien iskukuormitusten tarkasteluun

Research organisation: VTT
Project manager: Arja Saarenheimo, VTT

<table>
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<tr>
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<tr>
<td>Arja Saarenheimo, LicTech</td>
<td>VTT</td>
<td>Project manager, structural analyses</td>
</tr>
<tr>
<td>Kim Calonius, MScTech</td>
<td>VTT</td>
<td>Deputy project manager, structural analyses</td>
</tr>
<tr>
<td>Mikko Patalainen, MScTech</td>
<td>VTT</td>
<td>Structural analyses</td>
</tr>
<tr>
<td>Juha Kuutti, MScTech</td>
<td>VTT</td>
<td>Structural analyses</td>
</tr>
<tr>
<td>Ari Silde, MScTech</td>
<td>VTT</td>
<td>Liquid dispersal studies</td>
</tr>
<tr>
<td>Simo Hostikka, DrTech</td>
<td>VTT</td>
<td>Fire dynamic simulations</td>
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<tr>
<td>Topi Sikanen, MScTech</td>
<td>VTT</td>
<td>Fire dynamic simulations</td>
</tr>
<tr>
<td>Risto Lautkaski, LicTech</td>
<td>VTT</td>
<td>Explosion loads</td>
</tr>
<tr>
<td>Markku Tuomala, Prof</td>
<td>TUT</td>
<td>Analytical methods</td>
</tr>
</tbody>
</table>
Seismic safety of nuclear power plants – targets for research and education (SESA)
Ydinvoimaloiden seisminen turvallisuus

Research organisations: VTT, Institute of Seismology (Uni. Of Helsinki), Aalto University and AF-Consult
Project manager: Ludovic Fülöp, VTT

<table>
<thead>
<tr>
<th>Person</th>
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<tbody>
<tr>
<td>Ludovic Fülöp, PhD</td>
<td>VTT</td>
<td>Project manager; Coordination of the structural analysis, equipment qualification review, Teaching (SP1, SP2)</td>
</tr>
<tr>
<td>Vilho Jussila, MScTech</td>
<td>VTT</td>
<td>Modelling tasks of buildings in SP2</td>
</tr>
<tr>
<td>Jouni Saari, PhD</td>
<td>AF</td>
<td>Deputy project manager; Seismic hazard modelling methods – SP1</td>
</tr>
<tr>
<td>Marianne Malm, MSc</td>
<td>AF</td>
<td>Seismic hazard modelling – SP1</td>
</tr>
<tr>
<td>Niko Leso, MScTech</td>
<td>AF</td>
<td>Seismic hazard modelling – SP1</td>
</tr>
<tr>
<td>Pekka Heikkinen, Prof.</td>
<td>SeI</td>
<td>Coordination of hazard studies (SP1)</td>
</tr>
<tr>
<td>Päivi Mäntyniemi, PhD, docent</td>
<td>SeI</td>
<td>Teaching - Earthquake hazard assessment</td>
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<tr>
<td>Ilmari Smedberg, trainee</td>
<td>SeI</td>
<td>Earthquake hazard assessment – SP1</td>
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<tr>
<td>Timo Tiira, PhD</td>
<td>SeI</td>
<td>Data analysis of data-bank and developing the attestation relationships – SP1</td>
</tr>
<tr>
<td>Jari Puttonen, Prof.</td>
<td>Aalto</td>
<td>Overall planning of teaching activity, Teaching</td>
</tr>
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Extreme weather and nuclear power plants (EXWE)
Sään ääri-ilmöt ja ydinvoimalaitokset

Research organisation: FMI

<table>
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<tr>
<th>Person</th>
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<tbody>
<tr>
<td>Milla Johansson, MSc</td>
<td>FMI</td>
<td>Project manager, coordination, analysis and reporting, wind conditions and sea level variations.</td>
</tr>
<tr>
<td>Hilppa Gregow, MSc</td>
<td>FMI</td>
<td>Debuty Project manager and Project Manager, research on storm winds, freezing rain and meteotsunamis.</td>
</tr>
<tr>
<td>Kimmo Kahma, Prof</td>
<td>FMI</td>
<td>Supervising.</td>
</tr>
<tr>
<td>Kirsti Jylhä, Dr</td>
<td>FMI</td>
<td>Extreme temperature related investigations.</td>
</tr>
<tr>
<td>Kimmo Ruosteenoja, Dr</td>
<td>FMI</td>
<td>Climate change projections – processing of data.</td>
</tr>
<tr>
<td>Ari Venäläinen, Dr</td>
<td>FMI</td>
<td>Supervising</td>
</tr>
<tr>
<td>Matti Lahtinen, MSc</td>
<td>FMI</td>
<td>Extreme value analyses</td>
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<tr>
<td>Pauli Jokinen, MSc</td>
<td>FMI</td>
<td>Millenium model data analyses, weather patterns and extreme value analyses</td>
</tr>
<tr>
<td>Hilkka Pellikka, MSc</td>
<td>FMI</td>
<td>Meteotsunami related research, reviews and developing new methods for detection</td>
</tr>
<tr>
<td>Jenni Rauhala, MSc</td>
<td>FMI</td>
<td>Thunderstorms and meteotsunamis – resonance/propagation speed analyses by radar.</td>
</tr>
<tr>
<td>Juha Aalto, MSc</td>
<td>FMI</td>
<td>Testing possibilities in meteotsunami detection by statistical approaches and meteorological data.</td>
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<tr>
<td>Katri Leinonen, Trainee</td>
<td>FMI</td>
<td>Assessing the sea level scenarios</td>
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<tr>
<td>Hanna Boman, MSc</td>
<td>FMI</td>
<td>Meteotsunami related research</td>
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<tr>
<td>Markus Peura, Dr</td>
<td>FMI</td>
<td>Consulting in historical radar data retrieval</td>
</tr>
<tr>
<td>Seppo Saku, MSc</td>
<td>FMI</td>
<td>Freezing rain in Finland and Canada – case studies and literature review</td>
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<tr>
<td>Anu Karjalainen, Trainee</td>
<td>FMI</td>
<td>Meteotsunami related technical assistance</td>
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Risk assessment of large fire loads (LARGO)
Suurten palokuormien riskien arviointi

Research organisation: VTT
Project manager: Simo Hostikka, VTT

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<tr>
<th>Person</th>
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<tbody>
<tr>
<td>Simo Hostikka, DrTech</td>
<td>VTT</td>
<td>Project manager, Fire simulation model development and maintenance, OECD PRISME2</td>
</tr>
<tr>
<td>Terhi Kling, MScTech</td>
<td>VTT</td>
<td>Defence-in-depth</td>
</tr>
<tr>
<td>Johan Mangs, PhD</td>
<td>VTT</td>
<td>Power-Cable characterization for fire tests, smoke effects on digital automation</td>
</tr>
<tr>
<td>Topi Sikanen, MScTech</td>
<td>VTT</td>
<td>Simulations of liquid pool fires</td>
</tr>
<tr>
<td>Antti Paajanen, MSc</td>
<td>VTT</td>
<td>CFD-FEA interoperability, defence-in-depth</td>
</tr>
<tr>
<td>Anna Matala, MScTech</td>
<td>VTT</td>
<td>Simulations of cable fires</td>
</tr>
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PRA development and application (PRADA)
Todennäköisyyssopjaisten riskianalyysien kehittäminen ja soveltaminen

Research organisation: VTT
Project manager: Ilkka Männistö, VTT (-9/2012) and Ilkka Karanta (9/2012-)

<table>
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<tr>
<th>Person</th>
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<tbody>
<tr>
<td>Jan-Erik Holmberg, DTech</td>
<td>VTT</td>
<td>Human reliability analysis, integrated deterministic and probabilistic safety assessment (IDPSA)</td>
</tr>
<tr>
<td>Kristiina Hukki, MA</td>
<td>VTT</td>
<td>Human reliability analysis</td>
</tr>
<tr>
<td>Ilkka Karanta, LicTech</td>
<td>VTT</td>
<td>Project manager (from September 2012), IDPSA, level 3 probabilistic safety assessment</td>
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<tr>
<td>Ilona Lindholm, MScTech</td>
<td>VTT</td>
<td>MELCOR modelling</td>
</tr>
<tr>
<td>Ilkka Männistö, MScTech</td>
<td>VTT</td>
<td>Project manager (up to September 2012), EXAM-HRA participation, passive systems reliability analysis</td>
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<tr>
<td>Ahti Salo, DTech, professor</td>
<td>Aalto</td>
<td>Imprecise probability methods</td>
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<tr>
<td>Taneli Silvonen, MScTech</td>
<td>VTT</td>
<td>Passive systems reliability analysis, MELCOR modelling</td>
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<td>Antti Toppila, MScTech</td>
<td>Aalto</td>
<td>Imprecise probability methods</td>
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FinPSA knowledge transfer (FINPSA-TRANSFER)
FinPSA tietämyksen siirto

Research organisations: VTT and STUK
Project manager: Teemu Mätäsniem, VTT

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<tr>
<td>Teemu Mätäsniem, MScTech</td>
<td>VTT</td>
<td>Project manager, end user interviews, task control and reporting</td>
</tr>
<tr>
<td>Kim Björkman, MScTech</td>
<td>VTT</td>
<td>Deputy project manager, level 2 requirements</td>
</tr>
<tr>
<td>Tero Tyrväinen, MScTech</td>
<td>VTT</td>
<td>Quality assurance survey, test plan study</td>
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<tr>
<td>Ilkka Männistö, MScTech</td>
<td>VTT</td>
<td>End user interviews</td>
</tr>
<tr>
<td>Jan-Erik Holmberg, DrTech</td>
<td>VTT</td>
<td>Project manager (until May 2012), expert advisor</td>
</tr>
<tr>
<td>Ilkka Niemelä, MScTech</td>
<td>STUK</td>
<td>Level 1 requirements, software design re-documentation</td>
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Enhancement of Lappeenranta instrumentation of nuclear safety experiments (ELAINE)
Lappeenrannan ydinturvallisuuskokeiden mittausten ajanmukaistaminen

Research organisation: Lappeenranta University of Technology
Project manager: Antti Räsänen, Lappeenranta University of Technology

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<tr>
<td>Antti Räsänen, MScTech</td>
<td>LUT</td>
<td>Project manager, instrumentation, data acquisition, process control, experimental work</td>
</tr>
<tr>
<td>Vesa Riikonen, MScTech</td>
<td>LUT</td>
<td>Deputy project manager, experiment planning, analysis and reporting, data acquisition</td>
</tr>
<tr>
<td>Lauri Pyy, MScTech</td>
<td>LUT</td>
<td>PIV and WMS applications</td>
</tr>
<tr>
<td>Joonas Telkkä, MScTech</td>
<td>LUT</td>
<td>PIV and WMS applications</td>
</tr>
<tr>
<td>Harri Partanen, Engineer</td>
<td>LUT</td>
<td>Designing of test facilities</td>
</tr>
<tr>
<td>Hannu Pylkkö, Technician</td>
<td>LUT</td>
<td>Construction, operation and maintenance of test facilities, experimental work</td>
</tr>
<tr>
<td>Ilkka Saure, Technician</td>
<td>LUT</td>
<td>Construction, operation and maintenance of test facilities, experimental work</td>
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<tr>
<td>Eetu Kotro, Trainee</td>
<td>LUT</td>
<td>Construction, operation and maintenance of test facilities, experimental work</td>
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**Renewal of hot cell infrastructure (REHOT)**

*Kuumakammiotutkimusvaluksiin uudistaminen*

Research organisation: VTT
Project manager: Wade Karlsen, VTT

<table>
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<tr>
<th>Person</th>
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<tr>
<td>Wade Karlsen, Ph. D.</td>
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<td>Project manager</td>
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<tr>
<td>Päivi Karjalainen-Roikonen, MScTech</td>
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<td>Deputy project manager</td>
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<tr>
<td>Ulla Ehrnstén, MScTech</td>
<td>VTT</td>
<td>SEM and failure analysis topics</td>
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<tr>
<td>Seppo Tähtinen, MScTech</td>
<td>VTT</td>
<td>Hot laboratory design topics</td>
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<tr>
<td>Arto Kukkonen, Technician</td>
<td>VTT</td>
<td>Activated materials handling topics</td>
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<tr>
<td>Pentti Kauppinen, DrTech</td>
<td>VTT</td>
<td>Centre for Nuclear Safety design topics</td>
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<tr>
<td>Pekka Moilanen, DrTech</td>
<td>VTT</td>
<td>In-cell mechanical testing with bellows-driven devices</td>
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<tr>
<td>Tommi Kekki, MScTech</td>
<td>VTT</td>
<td>Centre for Nuclear Safety radiation safety</td>
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<tr>
<td>Tarja Jäppinen, MSc</td>
<td>VTT</td>
<td>In-cell NDE methods</td>
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<tr>
<td>Marketta Mattila, TechEng</td>
<td>VTT</td>
<td>Metallography and hardness testing of activated materials</td>
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<tr>
<td>Ulla Vuorinen, MSc</td>
<td>VTT</td>
<td>Radiochemistry and final repository materials testing topics</td>
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<tr>
<td>Timo Vanttola, DrTech</td>
<td>VTT</td>
<td>Centre for Nuclear Safety design topics</td>
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Administration of the Research programme (ADMIRE)
Tutkimusohjelman hallinnointi

Research organisation: VTT
Project manager: Kaisa Simola, VTT

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<tr>
<td>Kaisa Simola, DrTech</td>
<td>VTT</td>
<td>Project manager, programme administration, EU FP7 CCE-Fission Committee</td>
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<tr>
<td>Vesa Suolanen, MScTech</td>
<td>VTT</td>
<td>Deputy project manager, programme administration</td>
</tr>
<tr>
<td>Kari Rasilainen, DrTech</td>
<td>VTT</td>
<td>FP7 national support group</td>
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