SAFIR2014 Annual Report 2011

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 Institution 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 2011 was 9.5 M€ and 70 person years. Main funding organisations in 2011 were State Waste Management Fund VYR with 5.1 M€ and VTT with 2.9 M€. The programme has been divided into nine research areas and in 2011 research was carried out in 38 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.

Confidentiality Public

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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. An operating licence application will be made for the Olkiluoto 3 plant unit, and construction licence applications will be made for the Olkiluoto 4 plant unit and for the Fennovoima 1 plant unit. 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 that 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 2011 the realised volume of the SAFIR2014 programme was 9.5 M€ and 70 person years. Main funding organisations in 2011 were State Waste Management Fund VYR with 5.1 M€ and VTT with 2.9 M€. Research was carried out in 38 projects.

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.

Espoo 26.4.2012

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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 public call for research proposals was announced at the beginning of October 2010. 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 2011. The funding decisions were made by the State Waste Management Fund (VYR) in March 2011. In 2011 the programme consisted of 38 research projects and the programme administration.

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 2011 were 9.6 M€ and 9.5 M€ and 64 and 70 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 Institution 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 2011

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 2011, the research was performed in altogether 38 research projects. The total volume of the programme was 9.5 M€ and 70 person years. The research projects in the various areas with their planned and realised volumes are given in Table 2.1.

Table 2.1. SAFIR2014 projects in 2011.

<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>MANSCU</td>
<td>VTT</td>
<td>347</td>
<td>353.3</td>
<td>22</td>
<td>26.2</td>
</tr>
<tr>
<td></td>
<td>Sustainable and future oriented expertise</td>
<td>SAFEX2014</td>
<td>Aalto, TTL</td>
<td>110</td>
<td>131.6</td>
<td>10</td>
<td>14.9</td>
</tr>
<tr>
<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>178.5</td>
<td>15.1</td>
<td>15.0</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>222.3</td>
<td>15.5</td>
<td>15.5</td>
</tr>
<tr>
<td></td>
<td>Safety evaluation and reliability analysis of nuclear automation</td>
<td>SARANA</td>
<td>VTT, Aalto</td>
<td>436.6</td>
<td>454.8</td>
<td>36.3</td>
<td>36.3</td>
</tr>
<tr>
<td></td>
<td>Safety requirements specification and management in nuclear power plants</td>
<td>SAREMAN</td>
<td>VTT, Aalto</td>
<td>196</td>
<td>194.9</td>
<td>17.5</td>
<td>16.8</td>
</tr>
</tbody>
</table>
### Table 2.1. SAFIR2014 projects in 2011 (cont.).

<table>
<thead>
<tr>
<th></th>
<th>CRISTAL</th>
<th>VTT</th>
<th>298</th>
<th>293.9</th>
<th>21.5</th>
<th>24.6</th>
</tr>
</thead>
<tbody>
<tr>
<td>3.</td>
<td><strong>Criticality safety and transport methods in reactor analysis</strong></td>
<td>KAARME</td>
<td>VTT</td>
<td>230</td>
<td>221.7</td>
<td>16</td>
</tr>
<tr>
<td></td>
<td><strong>Three-dimensional reactor analyses</strong></td>
<td>KOURA</td>
<td>VTT</td>
<td>400</td>
<td>398.8</td>
<td>28.5</td>
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<tr>
<td></td>
<td><strong>Development of Finnish Monte Carlo reactor physics code</strong></td>
<td>NEPAL</td>
<td>Aalto</td>
<td>180</td>
<td>125.3</td>
<td>22</td>
</tr>
<tr>
<td></td>
<td><strong>Neutronics, nuclear fuel and burnup</strong></td>
<td>PALAMA</td>
<td>VTT</td>
<td>405</td>
<td>407.2</td>
<td>31.5</td>
</tr>
<tr>
<td></td>
<td><strong>Extensive fuel modelling</strong></td>
<td>BEPUE</td>
<td>VTT</td>
<td>94</td>
<td>94.0</td>
<td>6.5</td>
</tr>
<tr>
<td>4.</td>
<td><strong>Application of best estimate plus uncertainty evaluation method</strong></td>
<td>ESA</td>
<td>VTT</td>
<td>484</td>
<td>468.8</td>
<td>28.5</td>
</tr>
<tr>
<td></td>
<td><strong>Enhancement of safety evaluation tools</strong></td>
<td>EXCOP</td>
<td>LUT</td>
<td>259</td>
<td>258.8</td>
<td>19</td>
</tr>
<tr>
<td></td>
<td><strong>Experimental studies on containment phenomena</strong></td>
<td>NUFOAM</td>
<td>VTT, LUT, Aalto, Fortum</td>
<td>165</td>
<td>165.0</td>
<td>13.1</td>
</tr>
<tr>
<td></td>
<td><strong>OpenFOAM CFD-solver for nuclear safety related flow simulations</strong></td>
<td>NUMPOOL</td>
<td>VTT</td>
<td>120</td>
<td>119.4</td>
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<tr>
<td></td>
<td><strong>Numerical modelling of condensation pool</strong></td>
<td>PACSIM</td>
<td>LUT</td>
<td>50</td>
<td>50.6</td>
<td>4</td>
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<tr>
<td></td>
<td><strong>Improvement of PACTEL facility simulation environment</strong></td>
<td>PAX</td>
<td>LUT</td>
<td>383</td>
<td>386.2</td>
<td>27</td>
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<tr>
<td></td>
<td><strong>PWR PACTEL experiments</strong></td>
<td>SGEM</td>
<td>VTT</td>
<td>70</td>
<td>70.0</td>
<td>5</td>
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<tr>
<td>5.</td>
<td><strong>Modelling of pressure transients in steam generators</strong></td>
<td>COOLOCE</td>
<td>VTT</td>
<td>172</td>
<td>172.4</td>
<td>12.5</td>
</tr>
<tr>
<td></td>
<td><strong>Chemistry of fission products</strong></td>
<td>FISKE</td>
<td>VTT</td>
<td>189</td>
<td>182.1</td>
<td>11.5</td>
</tr>
<tr>
<td></td>
<td><strong>Thermal hydraulics of severe accidents</strong></td>
<td>TERMOSAN</td>
<td>VTT</td>
<td>299</td>
<td>203.4</td>
<td>21.9</td>
</tr>
<tr>
<td></td>
<td><strong>Transport and chemistry of fission products</strong></td>
<td>TRAFI</td>
<td>VTT</td>
<td>368.4</td>
<td>369.4</td>
<td>38.9</td>
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</table>
Table 2.1. SAFIR2014 projects in 2011 (cont.).

<table>
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<tr>
<th></th>
<th>Environmental influence on cracking susceptibility and ageing of nuclear materials</th>
<th>ENVIS</th>
<th>VTT</th>
<th>617</th>
<th>618.6</th>
<th>31</th>
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<td>6.</td>
<td>Fracture assessment of reactor circuit</td>
<td>FAR</td>
<td>VTT</td>
<td>255</td>
<td>255.1</td>
<td>14</td>
<td>17.8</td>
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<tr>
<td></td>
<td>Monitoring of the structural integrity of materials and components in reactor circuit</td>
<td>MAKOMON</td>
<td>VTT</td>
<td>238</td>
<td>238.2</td>
<td>15</td>
<td>17.4</td>
</tr>
<tr>
<td></td>
<td>RI-ISI analyses and inspection reliability of piping systems</td>
<td>RAIPSYS</td>
<td>VTT</td>
<td>155</td>
<td>155.1</td>
<td>11</td>
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<td></td>
<td>Advanced surveillance technique and embrittlement modelling</td>
<td>SURVIVE</td>
<td>VTT</td>
<td>118</td>
<td>118.3</td>
<td>7</td>
<td>7.0</td>
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<tr>
<td></td>
<td>Water chemistry and plant operating reliability</td>
<td>WAPA</td>
<td>VTT</td>
<td>197</td>
<td>182.2</td>
<td>18</td>
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<td>7.</td>
<td>Impact 2014</td>
<td>IMPACT2014</td>
<td>VTT</td>
<td>530</td>
<td>460.6</td>
<td>27</td>
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<td></td>
<td>Aging management of concrete structures in nuclear power plants</td>
<td>MANAGE</td>
<td>VTT, Aalto</td>
<td>224</td>
<td>227.1</td>
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<td>12.6</td>
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<td></td>
<td>Structural mechanics analyses of soft and hard impacts</td>
<td>SMASH</td>
<td>VTT</td>
<td>250</td>
<td>250.5</td>
<td>14</td>
<td>18.9</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>111</td>
<td>111.3</td>
<td>7</td>
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</tr>
<tr>
<td>8.</td>
<td>Extreme weather and nuclear power plants</td>
<td>EXWE</td>
<td>FMI</td>
<td>120</td>
<td>169.0</td>
<td>14</td>
<td>18.6</td>
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<td></td>
<td>Risk assessment of large fire loads</td>
<td>LARGO</td>
<td>VTT</td>
<td>207</td>
<td>210.0</td>
<td>12.8</td>
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<td></td>
<td>PRA development and application</td>
<td>PRADA</td>
<td>VTT, Aalto</td>
<td>232</td>
<td>237.4</td>
<td>18.7</td>
<td>18.7</td>
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<td>9.</td>
<td>Enhancement of Lappeenranta instrumentation of nuclear safety experiments</td>
<td>ELAINE</td>
<td>LUT</td>
<td>380</td>
<td>387.8</td>
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<td></td>
<td>Renewal of hot cell infrastructure</td>
<td>REHOT</td>
<td>VTT</td>
<td>154</td>
<td>154.0</td>
<td>9</td>
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<td>Programme administration*</td>
<td>ADMIRE</td>
<td>VTT</td>
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<td>9</td>
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</table>

*for period 1.1.2011-31.3.2012 with VAT 23% included

Summaries of research project results are given in the following subsections.
2.1 Man, organisation and human factors

In 2011 the research area "Man, organisation and human factors" consisted of two projects: Managing safety culture throughout the lifecycle of nuclear plants (MANSCU) and Sustainable and future oriented expertise (SAFEX2014).

2.1.1 Sustainable and future oriented expertise (SAFEX2014)

The purpose of SAFEX2014 is to generate new knowledge on evaluation methods of HR practices, knowledge and competence management in nuclear power industry and good practices in the area of human resource management, leadership and collaboration between organizations. In addition, the SAFEX2014 aims at creating practices for the development of leadership and management of human resources guaranteeing expertise and promoting collaboration of experts across organizational boundaries.

Specific goals in 2011

1 Human resource management in changing operational environment

The first subproject aims at identifying and developing methods and practices for human resource management in the nuclear energy industry. The focus is on methods and practices that support the work of HR-specialists and supervisors. The purpose of this task is to strengthen safe and reliable work and management practices.

1.1 Evaluation and benchmark of current human resource management and supervisory and leadership practices

This subtask involves a review of Human Performance Tools applied in the industry and benchmark study on good human resource management practices (especially in the context of nuclear power industry and high reliability organizations). In addition to the literature review, the data includes group interviews (one for HR-specialists and one for team/group leaders), which will be conducted in each participating organization in Finland. In addition, benchmark visit for example to Forsmark nuclear power plant will be conducted. During the visit some HR-specialists in Forsmark power plant are interviewed. Finally, IAEA documentation will be utilized to discover good human resource management practices in the global context of the nuclear energy industry. The deliverables of this subtask provide the input for the subtasks in years 2012-2014.

1.2 Development of human resource management and leadership practices

The research group’s findings regarding human resource management and leadership practices are summarized. This summary will be discussed and reviewed with the (HR) representatives of participating organizations and a development plan is prepared.

2 Expert collaboration across organizational boundaries and support for developing sustainable expertise

The second subproject aims at supporting the development and implementation of good collaboration practices (identified in previous SAFIR funded research projects) among nuclear power experts from different organizations. This subproject aims at supporting the development of sustainable expertise among nuclear power industry.
2.1 Improvement of collaboration between organizations in nuclear power industry

This subtask is built on the findings from previous SAFIR funded research projects concerning the factors that affect collaboration between organizations. The active role and contribution of participating organizations is utmost important for the success of this subtask. Development groups are established in each participating organization. In addition, one development group with representation from each participating organization is established. Groups define their goals, schedule, working methods, and topics for the expert collaboration crossing organizational boundaries. Communication between groups in different organizations will ensure shared or commonly agreed goals for future collaboration initiatives (carried out 2012-2014). The research group will assist in the establishment of the development groups as well as in the communication and collaboration within and between the development groups.

**Deliverables in 2011**

SAFEX2014 deliverables include two tangible outputs in 2011. First, a report on applicable, available, and feasible human resource management practices for nuclear power industry. Report findings are based on interviews and literature review. Second, a Master’s thesis on good leadership and supervisory practices on nuclear power industry.

At least following perspectives should be taken into account when developing assessment methods for nuclear power industry hr/supervisory/leadership work:

- It must be recognized that nuclear power industry organizations are complex and the work they do is complex as well

- Human resource management can be developed through different phases. These different phases offer organizations an opportunity to reflect and assess if they have achieved the targets of these different phases.

- Phase model recognizes the different roles and responsibilities of different actors

- An organization needs several different practices and methods for developing expertise and competences. The phase model can be used for assessing the feasibility of different methods and practices.

Both deliverables are intended to help organizations in nuclear power industry to develop their hr-practices and supervisory work. Findings are also utilized in workshops that are organized for Fortum, TVO, STUK, and Fennovoima. SAFEX2014 organizes two workshops in year 2012.

### 2.1.2 Managing safety culture throughout the lifecycle of nuclear plants (MANS CU)

The generic objective of the MANS CU project (2011-2014) is to develop safety management approaches with respect to three dimensions. Safety management approaches (such as rules and procedures, training, event investigation practices, safety culture evaluations and improvement programs, risk assessment and human performance programmes) should:

1) support the development of sufficient understanding and knowledge of nuclear safety and risks, as well as nuclear industry specific working practice demands. Research has shown that current safety practices rely strongly on the long experience and solid nuclear knowhow of the workers. However, nuclear safety related mechanisms may be hard to perceive and understand, even by the most experienced experts. Furthermore, efficient
communication of the nuclear industry related knowledge is important when the workforce generation change and new subcontractor companies are hired. Most current safety management practices emphasise attitude change and compliance. Those approaches need to be complemented with a clear focus on the actual context of the work, the nuclear safety related information.

2) take into account also the needs of other contexts than running plants. Safety management approaches should fit the purposes of design organisations, collaboration networks and supply chains, both in pre-operational and operational phases. The work and the management challenges e.g. in the design organisations or a construction project may vary from those of operating units. Thus, e.g. safety culture and human performance methods need rethinking. Furthermore, the supply chains of the power companies are likely to be increasingly multicultural and interdisciplinary in the future, which pose new demands for management.

3) support organisational alertness (mindfulness) to new risks which are based on either technical or social phenomena. When the operating units have a rather good performance records the culture may become over-reliant and complacent. Safety management tools need to support motivation for constant effort of continual improvement.

Specific goals in 2011

In 2011 the MANSCU project evaluates the use and usefulness of selected safety management concepts and approaches in different contexts. In 2011 MANSCU carried out three subprojects on topical safety management challenges. The subprojects were:

1) EVENTS: A review of the availability and usefulness of modern, systemic event investigation tools and software which could help the organisations to learn from safety culture in a way which takes the above mentioned three new challenges into account.

2) DESIGN: A study aiming for identifying the challenges of safety culture and human factors engineering in the design activities and in design organisations. The goal of the 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.

3) MOREMO: An analysis of actual working practices (both individual worker behaviours and organisational practices) in real outage situations at three nuclear plants. The aim is to support safety management and leadership with methods and models which take into account the fact that many activities are underspecified. New safety leadership needs to grasp how activities realize in the field, what is the rationale and possible safety effects of the behaviours.

In addition to these subprojects MANSCU continued the methodological work concerning safety culture, safety management and organisational evaluations. The aim was to develop further the DISC-model (Design for Integrated Safety Culture) and the theoretical premises about organisational safety by arranging internal workshops and publishing scientifically as well as through IAEA and other international forums.

Both the DESIGN and MOREMO subprojects aimed at integrating new concepts and methods to the safety culture methods developed by VTT in the earlier SAFIR2010-programme. In the DESIGN subproject the concept of HFE (Human Factors Engineering) was utilised in order to investigate how the design process might take human end-users into account. On the other hand the design organisations consist of human actors themselves and their safety culture affects, for
instance, their understanding of risks and prioritizing of goals. One of the tasks at 2011 was to evaluate how the Finnish design and regulatory oversight practices correspond with the international guidelines concerning HFE. Figure 2.1 illustrates the scope and stages of design where Human Factors Engineering could be employed. The smaller circle shows how HFE was mainly understood and utilised in Finland.

Figure 2.1. Currently the HFE is seen important when evaluating the human-technology interfaces, especially with respect to the new control room solutions. HFE, however, should be considered when other types of technologies are implemented. Furthermore, HFE activities do not need to focus on the evaluation stage, rather the users need to be considered much earlier in the design process.

The other subproject, MOREMO, aimed at testing methods with which to describe, evaluate and improve actual working practices in maintenance. The research integrates the organizational culture approach (CAOC) and resilience engineering. One of the methods tested in the project was FRAM (Functional Resonance Analysis Method). FRAM aims at identifying the functions of the activity and to characterize the functions and their interconnections. FRAM was tested in a diesel generator overhaul. With the FRAM model combined with knowledge of the working culture it is possible to anticipate the performance variability and its resonation in the system. Figure 2.1.2.2 is a FRAM model of the diesel engine overhaul. The Figure 2.1.2.2 illustrates how even a diesel service is a complex activity with numerous interdependencies between the different functions.

**Deliverables in 2011**

- EVENTS subproject reviewed two event investigation methods, SOL and ACCIMAP, which were selected on a basis of their systemic nature and focus on organizational, even societal aspects. Further, the FRAM method was described in the MOREMO report. It was discovered that no commercial Finnish or English language software is available of any of the methods. The working report illustrates the basic premises and steps of the methods.

- The DESIGN subproject reviewed the national and international guidance concerning Human Factors Engineering practices and compared the Finnish practices with the guidelines. The findings were reported in an international conference as well as in the intermediate report. The report also summarizes how the representatives of Fortum,
Fennovoima, STUK and Vattenfall perceived the safety challenges during design process and how they understood safety culture.

- Case studies at Loviisa, Ringhals and Oskarshamn power plants were carried out during the outages in order to test four different approaches in analysing working practises. The approaches were: Organisational Core Task modelling (OCT), FRAM (Functional Resonance Analysis Method), ETTO-principle (Efficiency-Thoroughness-Trade-Offs) and observations of working practices and culture. The case studies proved that actual work situations and practices can be best understood by combining the different methods. For that further research is needed. The preliminary findings are reported in a NKS report.

*Figure 2.1.2.2. The FRAM model of a diesel generator overhaul during an outage.*
A preliminary model of network safety culture cornerstones was created. The paper will be presented in PSAM/ESREL in Helsinki next summer.

Scientific articles were published or submitted on various MANSCU themes:

- human and organisational factors in the maintenance work
- developmental model of safety culture
- safety management biases
- inter-organisational relations

2.2 Automation and control room

In 2011 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)

The research project aims to improve the safety evaluation of I&C software in nuclear industry by improving consciousness of software process assessment and rationality of integrated evaluation methods. In 2011, the following tasks were carried out: 1) Development S4N assessment method and process, 2). Evaluations coverage and rationality, 3) Reading techniques, and 4) Novel technologies.

Specific goals in 2011

This task of the project Coverage and rationality of the software I&C safety assurance (CORSICA) aims at developing an assessment method and process called Nuclear SPICE (S4N). It defines a set of development processes and their capability scale. The results of an assessment can be used for the qualification of safety critical I&C systems and software. A highly capable process and conformance with standards are not alone an absolute guarantee for safety. Safety can be achieved only by using several different approaches, which all provide their own evidences and support for qualification and licensing. Nuclear SPICE is a step towards such an integrated approach. In 2011 our focus was in the core elements of Nuclear SPICE.

Nuclear SPICE assessment method and process is adaptable for different qualification needs. It can be used at different levels of rigor. Tailoring feature is required for practical reasons, because real-life situations vary a lot in qualification. The assessment process defines also the minimum requirements for assessment sufficiency. The most important topic is the degree of coverage and completeness of different evidences and evidence types.

Nuclear SPICE will be developed further in the coming years, to cover strict safety requirements. Results will also be used to evaluate the reliability of the target system and software. Moreover, conformance with standards will be improved, especially with SIL4
requirements of IEC 61508 and Class A in IEC 61513. It is also possible that some additional models from regulatory bodies will be integrated in Nuclear SPICE in the future.

![Figure 2.2.1.1. Layers of taxonomy in Nuclear SPICE.](image)

**Task: Evaluations coverage and rationality**

Quality of system and software engineering artefacts is related to effective and complete verification and validation. The task aims to prepare a set of applied methods and techniques commonly accepted and used in I&C design and implementation in NPP at the highest safety class. This task will give answers to the question “Out of everything that is possibly to evaluate, which features are the right ones to take?” An initial list of factors is the following: credibility of applied methods, efficiency, accuracy and precision, and performance. Especially, concentration in 2011 is on functional testing as a rational process and achievement of common understanding of proper methods and techniques to be used in evaluation of I&C software artefacts in safety Category A.

**Task: Reading techniques**

Reading techniques are step-by-step procedures that provide guidance while software documents are inspected. The techniques are typically well-defined ways of inspecting documents, and they can be customized to suit a particular organization. The most common reading techniques are simple ad-hoc reading and a checklist-based reading technique. However, more advanced and detailed procedures have been created for various purposes. Benefits of using these advanced reading techniques include increase in the amount of faults found, and lower inspection costs.

The objective of this task is to provide an overview of different reading techniques, and their application to the inspection of various software documents. The focus of this task is on a specific reading technique called Perspective-Based Reading (PBR). The goal is to find out how this particular technique can be applied to the inspection of nuclear domain software I&C documents.
Task: Novel technologies

The task aims to survey new technologies that have been or are likely to be introduced into NPPs in the near future. The technologies currently in use have problems that generate an interest to find alternatives as new NPPs are constructed and old ones are renewed and modernised. Technologies used in automation systems with less strict safety requirements can mature to provide sufficient reliability, along with technology specific advantages, to make them interesting also for safety IC system perspective. On one hand, the technology itself is considered as well as the currently existing systems offered for use in NPPs. On the other hand, the prevailing situation in licensing and policies of authorities is considered. The results are documented in technology specific reports in VTT’s publication series and presented in the Ad Hoc meetings. In 2011 the goal is to survey FPGA (Field Programmable Gate Array)-technology.

Deliverables in 2011

- The task “Development of S4N assessment method and process” in 2011 was to define a process assessment based approach to ensure quality in software development for nuclear domain.

- The FISMA report 2011-1 titled “S4N method description - Nuclear SPICE PRM and PAM” defines a model called Nuclear SPICE. We explain how it combines both SPICE related generic requirements with functional safety and nuclear specific elements. Nuclear SPICE contains two models: process reference model (PRM) and process assessment model (PAM). Main content of the report is the list and definition of processes in the PRM. The processes cover system and software development, technical support processes, project management and safety management. Some processes are added to cover requirements in IEC61513 and IEC60880 standards.

- The FiSMA report 2011-2 titled “S4N assessment process - Requirements for Nuclear SPICE assessment” describes requirements and activities for performing systems and software process assessment in highly safety-critical environments in nuclear industry domain. The requirements will be further developed in 2012, and the assessment process will also be validated. This approach relies on ISO/IEC 15504 process assessment standard. The standard based requirements are considered mandatory - rest of the requirements can be prioritized. The document supports the use of Nuclear SPICE Process Assessment Model (PAM) that is described in FiSMA report 2011-1, which presents the applicable process descriptions and assessment indicators. This report concentrates on the issues related to the conduct of an assessment.

- Two concepts are under discussion in the report “Rationality of functional testing of Category A software”: coverage and rationality. The focus of the discussion is on functional testing of safety-critical software of instrumentation and control in nuclear power plants. The safety level of the software and corresponding computer system is Safety Class 2 according to Finnish YVL Safety Guides and Category A according to IEC 61226.

- In practice, the rationality of functional testing is worked out by applying test cases produced with equivalence partition testing connected to boundary value analysis. The theoretical bases for both of these methods are inescrutable. Therefore, more conditions must be set for inspections of not only functional testing, but all review and inspection techniques that provide high-quality testing. In addition, the credibility of argumentation on the satisfaction of test results is an important issue. Setting rigour grades for methods and techniques aids
developers and users in their duty when qualifying application software for safety-critical I&C systems in NPPs.

- The report “Application of the Perspective-based Reading Technique in the Nuclear I&C Context” reviews the state-of-the-art software reading techniques used in inspections and reviews, and briefly reviews some of the empirical research in this context. The majority of the empirical research results indicate that, for example, perspective-based reading is more cost-effective and can detect more defects than more basic reading techniques.

- We have studied the perspective-based reading technique closely, and applied the technique to the inspection of nuclear-domain requirement specifications. First, we identified the nuclear specific concerns that needed to be taken into account in the nuclear domain. The identified special concerns were: the complexity of the design process, the long system life-span, regulator’s role in system development, sub-contractor issues, and the great importance of safety. Then we designed seven perspective-based reading scenarios.

- The report “The Current State of FPGA Technology in the Nuclear Domain” presents an overview of FPGA technology, including hardware aspects, the application development process, risks and advantages of the technology, and introduces some of the current systems.

- Field programmable gate arrays are a form of programmable electronic device used in various applications including automation systems. In recent years, there has been a growing interest in the use of FPGA-based systems also for safety automation of nuclear power plants. The interest is driven by the need for reliable new alternatives to replace, on one hand, the aging technology currently in use and, on the other hand, microprocessor and software-based systems, which are seen as overly complex from the safety evaluation point of view. Currently the number of FPGA-based applications used for safety functions of nuclear power plants is rather limited, but it is growing.

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

The project focuses on studying how humans and automation systems collaborate to accomplish safety and production goals of nuclear power plants (NPPs). Specifically, the project aims at (1) gathering knowledge of procedure and human-system interface (HSI) solutions supporting accident management; (2) providing knowledge of the effect of the new automation concept on operator behaviour in incident and accident situations; (3) identifying Human Factors Engineering (HFE) considerations important in the selection of a control room (CR) modernization strategy; (4) investigating the effect of the operators’ automation awareness and their automation skills on operator behaviour in incident and accident situations; and (5) developing methods and tools for the training of digital automation. Practical methods and guidelines for the design of procedures and HSIs supporting accident management are developed based on empirical research. Knowledge on factors affecting automation awareness and automation skills, and methods for their development are also provided. As results of this work, a detailed Concept of Operations (ConOps) for emergency situations can be developed.

The aim is also to develop expertise on the accomplishment of HFE activities during design process, further promote international collaboration with research and expert organizations and institutions and strengthen delivery of expertise in Finland in the field of user-centred design of NPP CR systems.
Specific goals in 2011

In 2011, the research in the project has focussed on five main areas: First, it has been studied the effect of emergency operating procedures (EOPs) on the operators’ situation awareness in different phases of the accident management process and the effect of routines of procedure usage on the management of severe accidents. A literature review on procedure usage has been prepared based on previous empirical work and the use of EOPs has been studied based on empirical simulator data gathered in the previous SAFIR/O’PRACTICE-project. Second, it has been studied the effect of digital I&C systems on operator practices in accident situations by preparing a literature review on operator practices in accident management in hybrid control rooms. Third, it has been identified challenges in the design of interactive large screen displays (LSDs) for the simulator environments by interviewing the designers of an interactive virtual panel system. Fourth, it has been prepared a literature review on operators’ automation awareness and on the effect of automation complexity on automation awareness and automation skills. Fifth, the concept for an Apros-based simulation environment has been outlined.

Deliverables in 2011

- In an interview study, operators’ conceptions on procedures, structured under three themes relevant to the role of the operator’s autonomy in relation to the official guidance, have been studied. The interviews were analysed by classifying each interview answer into one of three predefined categories: reactive, confirmative or interpretative orientation. The results show striking similarities in the orientations of the operators of two different NPPs. Interestingly, the orientation of turbine operators seems to differentiate from the ones of the other operators (shift supervisor and reactor operator).

- The operators’ procedure usage has been studied from the perspective of three tool functions: instrumental, psychological, communicative. In a literature review, two hypotheses were developed for the analysis of the results of empirical studies. The first hypothesis is connected to the psychological and the second to the communicative tool function. The formulation of the hypotheses is partly based on the results and ideas of Filippi (2006) and Dien (1998). According to the first hypothesis, too strong level of guidance restricts adaptive operator activity, and according to the second one, too strict division of tasks between operators deteriorates crew’s shared understanding of the situation.

- It has been studied operator practices in accident management in wholly or in partly digitalized control rooms. Based on the literature review, it was found that differences in operator performance are quite small between traditional analogue and digital control rooms in design based accident situations. Even though computer-based human-system interfaces and new techniques in process control open new possibilities for process control, a general finding is that in traditional analogue control rooms situation awareness at the team level is higher and communication more fluent than in digital control rooms.

- Designers of the Fortum Virtual panel system were interviewed in summer 2011. The virtual panel concept is realized as a part of the training simulator’s human system interface by introducing a set of touch sensitive wall mounted displays for the control room of the training simulator. Based on an interview with the designers and an interface walkthrough, it was addressed what kind of challenges and compromises the designers needed to account for in the transition from the analogue to the digital interface medium, and how they envision the interactive surface technologies could be used in the future control room environments.
A descriptive model of the innovative features of the Virtual panel concept within the System Usability framework was developed (see Table 2.2.2.1).

- A literature review on automation awareness and its development was prepared. Human-automation interaction can be viewed from different perspectives. Dimensions that were used in this review were levels, functions, flexibility, modes, reliability and processes (see also O’Hara et al., 2010). A preliminary model on automation awareness has been developed (see Figure 2.2.2.1). According to our view, the complexity of the automation system has a great impact on the development and maintaining of automation awareness. Other factors that have to be taken into account are the usability and functionality of human-system interfaces, personnel’s trust in automation and various situation-specific factors.

- The development of an Apros-based simulator environment that will be used in the experimental testing of automation awareness has been started.

*Figure 2.2.2.1. Conceptual model of automation awareness.*
Table 2.2.2.1. Description of the innovative features of the Fortum Virtual panel system

<table>
<thead>
<tr>
<th>Core task demands</th>
<th>Readiness to act</th>
<th>Flexibility of acting and reorientation</th>
<th>Interpretativeness of acting</th>
<th>Conceptual mastery and comprehending wholes</th>
<th>Creating shared awareness</th>
<th>Optimal sharing of efforts</th>
</tr>
</thead>
<tbody>
<tr>
<td>HSI concept requirement</td>
<td>Training System enables operators to learn meaningful and appropriate control of process by adhering to the conventions of the established HSI solution</td>
<td>Process control System enables the control of the process phenomena by providing the same control possibilities as the conventional panel system</td>
<td>Process monitoring System enables the focus on relevant process events by presenting the process in the same fashion as the conventional panel system</td>
<td>Communication and cooperation System enables operators to form shared understanding about the process and available resources by providing a common reference</td>
<td></td>
<td></td>
</tr>
<tr>
<td>HSI concept solution</td>
<td>Digital photographs of the conventional analogue control panels (as a background) to replicate the look of the established HSI solution</td>
<td>Touchscreen technology for creating an interactive surface for executing the control operations</td>
<td>Digital controls and status information integrated on the photos of the conventional control panels to create an interactive Virtual panel system</td>
<td>The Virtual panel system are installed in accordance with the layout of the conventional wall-mounted panels and bench boards</td>
<td>Traditional communication channels to create common ground</td>
<td></td>
</tr>
<tr>
<td>Interface requirement</td>
<td>Visual cues are consistent with the way of accomplishing control operations</td>
<td>Freedom for accomplishing control operations</td>
<td>Enables continuous operation</td>
<td>Enables multiple operations simultaneously</td>
<td>Provides feedback for the operations</td>
<td>Provides visibility for distant monitoring</td>
</tr>
<tr>
<td>Interface solution</td>
<td>Presents the process mimic</td>
<td>Introduction of double operation mode</td>
<td>Touch sensitive control elements</td>
<td>(Audio) visual operation feedback</td>
<td>Presentation of status information</td>
<td>Means for sharing information</td>
</tr>
<tr>
<td>Tool functions</td>
<td>Instrumental function</td>
<td>Psychological functions</td>
<td>Communicative functions</td>
<td></td>
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<tr>
<td></td>
<td>Electricity switch</td>
<td>Valve control element</td>
<td>Dynamic colour change</td>
<td>State indicator lights</td>
<td>Information tags</td>
<td></td>
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<tr>
<td></td>
<td>Pump control element</td>
<td>Electricity switch</td>
<td>Capacity controller</td>
<td>Fault indicator lights</td>
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<td></td>
<td>Sipart controller</td>
<td></td>
<td></td>
<td>Measurement indicator</td>
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</table>
2.2.3 Safety evaluation and reliability analysis of nuclear automation (SARANA)

The general objective of the whole four-year project is to develop methods and tools for safety and reliability analysis of digital systems and utilize them in practical case studies. The project provides guidelines to analyse and model digital systems in PRA context, brings together deterministic and probabilistic analyses for safety assessment of plant designs, develops the reliability analysis tool YADRAT, investigates the applicability of probabilistic model checkers, uses model checking for analysing high-level design architecture, develops an iterative and automatic algorithm for modular model checking of large systems, develops methods for systematic model checking of complex asynchronous systems, and investigates how different modelling approaches and tools can be used for the safety analysis of I&C in different phases of the plant lifecycle. The project follows and participates in the work of several international working groups and conferences in its application areas.

Specific goals in 2011

- Make a state-of-the-art review on integrated use of DSA (deterministic safety assessment) and PSA (probabilistic safety assessment) to be discussed in a Nordic workshop jointly organised with KTH and ScandPower.

- Determine how model checking could be used in the analysis of high-level architecture descriptions and to develop methodology to model various types of faults on the architecture level.

- Perform a pre-study on how different modelling approaches and tools are used and suited for the safety and reliability analysis of I&C during the different phases of the lifecycle.

- Develop a fictive example of digital I&C system of a nuclear power plant and use it to provide guidelines to analyse and model digital systems in PRA context.

- Explore what type of methods presented in the IEC 61508 could be applicable in the PSA domain. Organise a Nordic workshop and establish a European network and contacts with related project and activities.

- Make a state-of-the-art pre-study report on probabilistic model checking tools and investigate how model checking models can be enriched with probabilistic information.

- Study aspects of reliability theory in the domain of DFM (dynamic flowgraph methodology) and BDD (binary decision diagram) modelling. Compare the computational efficiency and scalability of two computation algorithms based on BDD and to study new approaches to support BDD-based algorithms to increase the scalability of the reliability analysis tool YADRAT.

- Develop an automatic modular model checking algorithm to iteratively search for a composition of modules that at the same time is computationally manageable, and covers enough modules to prove the fulfilment of temporal properties in the original model.

- Develop methods that allow systems operating with their own clocks to be modelled accurately and to explicitly model the delays inducing data transmission interfaces between subsystems. Find fault scenarios that are difficult to reproduce.
Deliverables in 2011

- A Nordic workshop organised on the integrated use of DSA and PSA. An international network on this topic was established aiming e.g. to prepare a joint project proposal for the Euratom FP7 2012 Call.

- Feasibility of model checking for the analysis of high-level architecture descriptions investigated through a case study based on an imaginary example system. The case study consisted of creating a combined model for the system’s logical function, hardware configuration, and failure modes of hardware components based on typical FMEA information. The model allowed the verification of system properties under certain failure assumptions (e.g. single-failures).

- A pre-study report on applicability of different reliability and safety modelling and evaluation techniques was finished. The report describes how different methods and techniques such as PRA, simulation, model checking, code analysis suit for analysing software systems for nuclear power plants and for which life cycle phases they suit best.

- Several meetings of the WGRISK task group organised. Two public workshops held in Washington D.C. and in Espoo. Failure modes taxonomies collected, draft of the fictive digital I&C system example created, and draft list of contents of the guidelines prepared. Interim report NKS-261.

![Figure 2.2.3.1. Example of a digital I&C protection system.](image)

- Use of IEC 61508 in nuclear applications regarding software reliability was investigated. The use of SILs (safety integrity levels) to determine the failure probability of the software used in NPP applications was found to be a controversial issue. A rough method for determining the failure probability of software was outlined for further development.

- State-of-the-art review on probabilistic model checking methods and tools has been completed. The Prism model checker was found to have many features useful for the analysis of both discrete and continuous time Markov chains as well as Markov decision processes. The scale of the models handled by the tools is significantly smaller than traditional model checking tools and the methods to integrate the use of these tools with other validation methods is a topic of further research.

- The computational efficiency of two BDD based algorithms was studied and the one that performed better was selected for further development for the reliability analysis tool YADRAT.
- The main algorithm for model checking large systems was established and implemented to work together with the NuSMV model checking tool. The modular algorithm is capable of iteratively searching 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.

- A new modelling approach was developed for Uppaal model checker. The new approach significantly enhances the scalability of the Uppaal models using a combination of BDDs (for model preprocessing) and explicit state timed model checking (by Uppaal) approaches.

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

Unambiguous requirements combined with systematic configuration management and traceability are a key to ensuring plant safety and economy. Requirements are also necessary for seamless communication in large, geographically distributed project organizations. The aim of the SAREMAN research project is to develop good practices for requirements specification and management in nuclear power plants. The focus is on safety but also other kinds of requirements related e.g. plant availability are concerned. This effort is founded on a structured conceptual model of requirements and requirements engineering, as well as the major elements of a nuclear power plant. SAREMAN is planned to be a four-year project 2011-2014. 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. In this work, the project team will make use of existing international standards and recommendations. In order to limit the scope SAREMAN focuses on monitoring and control of plant functions and systems. This highlights the multi-disciplinary nature of requirements engineering involving, for example, safety engineering, process engineering, I&C systems and control room operations. The principles of requirements engineering are, however, common to several application areas. So, the results of the project, consisting of precise terminology, document templates, guidelines and ICT tool demonstrations will be widely applicable to the design and licensing of nuclear power plants.

![Figure 2.2.4.1. Overall work plan for the four-year period and the focus in 2011.](image-url)
Specific goals in 2011

In 2011, the focus of the SAREMAN project was on making the terminology more precise and collecting recommendations for unambiguous requirement specifications, both tailored for the NPP domain. This work was concretized with sample requirement specifications of a suitable safety-related plant system. The research activities were organized into the following tasks:

1. **Conceptual model of requirements engineering**: The entities needed in requirements engineering in NPP, their attributes and relationships were defined on the basis of existing standards, guidelines and current practices. In addition to textual definitions, the concepts were visualized with UML (Unified Modeling Language) diagrams. Also a more traditional glossary was collected with some new interpretations of relevant terms.

2. **Readability of design artefacts**: A limited pre-study was carried on the readability of regulatory guidelines and design specifications. This topic was considered relevant because readability is a quality attribute of design artefacts important both for traceable V&V and for communication among design parties. However, as not immediate uses of readability analysis tools were not foreseen, the work was limited to a literature review. The observations were documented as set of slides and used as an input to the development of requirements documentation recommendation in the other tasks.

3. **Documentation methods and templates**: This task was planned to review existing recommendations and standards for expressing requirements and writing high-quality requirements specifications. In 2011 the focus was on textual requirements and on two specific topics: structured natural language requirements and the Concept of Operations (ConOps). The first topic worked with a method called Easy Approach to Requirements Syntax (EARS). It was applied to YVL guidelines and an EPRI report on rod control requirements. The findings were documented in the form of training material. Concerning the second topic, a working report on the idea, content and development of a ConOps was written. Templates for documenting Concept of Operations and System Requirements were outlined and applied in the rod control example in task 4.

4. **Sample requirement documents**: The aim of this task was to demonstrate and evaluate the results produced in the previous tasks. Reactivity control function and rod control system were agreed as the topic and sample documents on the Concept of Operations and System Requirements were written on the basis of available material and the principles from tasks 1 and 3.

5. **Reporting and project management**: In addition to managerial issues, this task was allocated for finalising and disseminating the results from other tasks. Three conference papers were written and presented during the first project year.

Deliverables in 2011

- **Conceptual model for safety requirements specification and management in nuclear power plants**: The goal of this working report is to provide clear terminology as a foundation for design. One of its starting points is that requirements cannot be discussed in isolation from other engineering activities and system descriptions. Therefore, this report is not limited to requirements but discusses the principles of modelling complex socio-technical systems in a broader sense. A second starting point is that the number of requirements and dependencies requires computer tools. Computer tools, in turn, need formal data models.
This is why this working report has taken influences from international standardisation of product data modelling. In order to support current practices more directly, an appendix gives common-sense definitions of key terms.

- **Readability of safety-related documents in the nuclear field – a pre-study:** This set of slides collects the findings from the small literature review carried out. The main conclusion was that the clarity of safety requirements and technical documents in general is important for V&V and communication in large networked projects. The mission of SAREMAN is to improve the quality of requirements specifications, including their readability as one quality attribute. Readability is an interesting but wide research area. So, it is not explicitly included as a task in the plan for 2012. However, it will be considered as one aspect in high-quality requirements documentation.

- **Training material for specifying requirements in structured natural language:** This training material consists of a tutorial presentation, slides, and supporting material on the Easy Approach to Requirements Syntax (EARS) method. It also presents briefly other good requirements engineering guidelines. The training material aims to support practitioners in applying EARS to requirements specifications. The tutorial has been piloted by two Master’s thesis workers at Aalto University and presented to a group of key stakeholders in STUK. The method has been seen as useful, and work on this area is continued in 2012.

![Diagram](Figure 2.2.4.2. Examples from the SAREMAN conceptual model.)
• **Concept of operations (ConOps) in the design of nuclear power plant instrumentation & control systems:** Complex systems contain both technical and human elements, and their multi-disciplinary design needs shared modelling methods. The idea of Concept of Operations (ConOps), included in many guidelines and standards, could be used as such an integrative element. The aim of this working report is to interpret the general ideas to the design of I&C systems in nuclear power plants. The report discusses modelling concepts, representation methods and working practices for developing a ConOps. The role of operational scenarios is to gather stakeholder needs, to foster communication and to test the conceptual solutions made.

• **Sample requirement specifications:** This deliverable includes two memos on an imaginary upgrade of a control rod control system. The first one, “Concept of Operations for the control of reactivity Upgrade of the rod control system, an example”, provides a small example of a ConOps specification that is not stored in a computer-based design database but written as a “traditional” document. It is refined in the memo titled “System requirements specification for control rod control system upgrade – an example”. The aim of these two documents is to illustrate the general principles suggested in the other working reports.

### 2.3 Fuel research and reactor analysis

In 2011 the research area "Fuel Research and Reactor Analysis" consisted of five 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) and Extensive fuel modelling (PALAMA).

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

The knowledge on various calculation methods must be maintained and improved. The reactor physics code suite has to cover both normal and transient or accident conditions. Moreover, the expertise in the field has to be improved as a lot of knowledge has been lost recently and will be lost in the future. Criticality safety is an increasingly important subject where more experts are needed. Determination of the uncertainty of various parameters and their importance for the final results is necessary in order to understand the accuracy of computer programs. These methods will be developed within this project.

**Specific goals in 2011**

Specific goals in 2011 included a lot of training of young researchers in order to improve the expertise. One person was to be trained to make inventory calculations and at least two persons into criticality safety studies.

The UAM benchmark continued in 2011. The updated reactor physics exercise was computed with both CASMO-4 and SCALE 6.1. In addition burnup methods were studied. The ultimate goal is to enable the propagation of an uncertainty estimate through the whole burnup calculation chain. During 2011 the research concentrated on the error related directly to the solving of the burnup equations.

In criticality safety the main emphasis was in education and knowledge management. Three persons started to familiarise themselves with criticality safety standards and analysis
methods. A strong focus was on 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.

Figure 2.3.1.1. The moderator density affects strongly the multiplication factor of the VVER-440 assembly. It is clearly seen that the assembly changes from under moderated to over moderated when the moderator density exceeds 0.2 g/cm³.

**Deliverables in 2011**

- Basic knowledge of reactor physics was enhanced through a study of cross section dependences on reactor physics parameter. The study was conducted using an EPR 4.9% enriched assembly. The group constants were calculated with CASMO-4E for seven burnup points between 0 and 50 MWd/kgU. For each reactor physics parameter five points were used. The data was then compared to fits used by HEXBU-3D and TRAB-3D. The group constants behaved in general almost linearly and the fits used by the codes seem to be reasonably good.

- The expertise in deterministic transport and inventory calculations was increased through a study in the OL1/2 components outside the reactor core using the DOORS package. In addition this package was used in a VVER-440 Final Disposal CB 6 Benchmark exercise. The model for a disposal cask was created for TORT calculations. The decay calculations for activities of 1 million years and 113 materials were performed.

- The updated reactor physics exercise in the UAM benchmark was calculated with both SCALE6.1 and a modified version of CASMO-4 that containing generalised perturbation theory (GPT). This modification allows performing S/U analysis for other responses in addition to the multiplication factor. The agreement between the two codes was good giving credence in the developed methodology. However, it is noteworthy, that the modified CASMO is remarkably faster in GPT calculations than SCALE 6.1.

- The knowledge in criticality safety issues was increased by a study on parameter dependences. All the geometrical parameters of a VVER-440 assembly were varied and the de-
pendency of the multiplication factor on each parameter was studied systematically. In addition, a few material parameters were also included in the study. The study was done with both MCNP and Serpent, and with two cross section libraries ENDF/B-VII and JEFF-3.1.1. This study gives a good understanding of the effect of various assembly level parameters on the reactivity.

- The effort to create a criticality validation package for the VVER-440 for reactor physics codes used at VTT for criticality safety analyses has been restarted. The existing MCNP input files for the package were revised to agree with the benchmarks in the criticality safety handbook. The package comprises at the moment 23 critical lattices. The ultimate goal is to have 100 to 200 lattices in the package.

- An ad hoc group on criticality safety was formed. This group was warmly welcomed as an informal forum where issues related to various criticality safety related issues can be discussed. There is a clear need for this kind of a forum in Finland. Each power company and VTT has a member assigned to the group. Additional people have been invited to the meetings depending on the topics to be discussed.

2.3.2 Three-dimensional reactor analyses (KOURA)

The goal of the project is to have truly independent transient calculation system, which can be utilized by the safety authority and other end-users for safety analyses. To achieve this, the 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 2011

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 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 HEXTRAN, 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.
Figure 2.3.2.1. On the left liquid temperature in the EPR reactor pressure vessel, on a vertical plane through the centreline. On the right unstructured mesh for the EPR pressure vessel.

Deliverables in 2011

- PORFLO was rewritten during 2011. Two major improvements in the new version are the ability to handle unstructured meshes and parallelization. The ability to handle unstructured meshes enables the use of structure fitted meshes, which is major improvement when simulating cylindrical, or otherwise rounded, shapes such as reactor pressure vessel. New version utilizes general CGNS format that improves possibilities to comparisons to other CFD codes. Report of the new PORFLO development has been written.

- PORFLO calculations were continued with the EPR reactor pressure vessel model that was developed in TRICOT project in SAFIR2010 programme. The older reactor pressure vessel model uses rectilinear meshes. New model utilizing structure fitted meshes was developed for EPR reactor pressure vessel.

- New version of the PORFLO code has been applied also to a PWR fuel assembly. Applications of PORFLO have been described in the research report and in the conference paper, which has been submitted to the ICONE-20 conference.

- TRAB-3D input files for several BWR transients have been updated to correspond to the present program version. Transients have been recalculated with TRAB-3D. Internal
coupling of TRAB-3D/SMABRE has been improved for BWR transients. Work has been reported.

- International co-operation has been continued in the framework of AER and OECD/NEA. Participation in the meetings of NEA Nuclear Science Committee and Working Party on the Scientific Issues of Reactor Systems (WPRS), as well as the AER symposium and AER working group D meetings has been part of the project.

- OECD/NEA launched the BWR stability benchmark O2. VTT participates in the benchmark with TRAB-3D. Work with the TRAB-3D benchmark input has been started. Status report of VTT’s participation has been written.

- VTT participates in the 7th dynamic AER benchmark concerning re-connection of an isolated circulation loop in a VVER-440. An Apros model utilizing 3d flow model and 3d nodal neutronics model has been prepared for VVER-440/213. Test calculations have been performed and status of the modelling has been reported.

![Figure 2.3.2.2. Testing of VVER-440 reactor pressure vessel model prepared for the 7th AER benchmark: water temperature in reactor pressure vessel.](image)

- Manual for hot channel analysis with TRAB-1D has been prepared. Reactor dynamic codes TRAB-3D and HEXTRAN have been transferred to VTT’s Linux cluster and modification of their source code to Fortran90 standard has been started.

- One researcher participated to the FJOH summer school.
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 about 160 registered users in 68 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 40 users with previous experience in Serpent 1. The project has close collaboration with the CRISTAL and NEPAL projects in SAFIR2014.

Specific goals in 2011

Re-writing of the Serpent source code started in September 2010, and most of the efforts in the KÄÄRME project are devoted in the development of Serpent 2. For year 2011 this includes completing the main functionality required for running the transport simulation, with capability to perform group constant generation and burnup calculation. Another major goal is to better adapt Serpent to parallel calculation and super-computing, by focusing on certain fundamental flaws in memory management and the structure of the source code. Multi-physics applications, involving coupling of Monte Carlo neutronics to thermal-hydraulics and fuel performance codes will be a major topic later in the project, and the development of some related features was already started in 2011.

Deliverables in 2011

- The source code was completely re-written, and structured in a way that better accounts for coupling the transport routine into the depletion solver. Most of the main capabilities in Serpent 1, including group constant generation and built-in burnup calculation routines were implemented in Serpent 2.

- Problems related to excessive memory usage in Serpent 1 were overcome in Serpent 2 by introducing different optimization modes for small and large-scale problems. This extends the burnup calculation capability from 2D assembly-level problems to full-core calculations involving tens or hundreds of thousands of depletion zones. The optimization modes are documented in a conference paper accepted in PHYSOR-2012.

- Parallelization routines in Serpent 2 were completely re-written, with a new approach based on hybrid MPI/OpenMP methodology. The new approach enables running the code in parallel mode in large multi-CPU clusters without limitations. The work also involved the development of a new 64-bit linear congruential random number generator, which enables the reproduction of the same random number sequence in both serial and parallel mode.
Figure 2.3.3.1. Neutron density distribution in a BWR fuel assembly. Calculated using Serpent 2.

- Development of a gamma transport routine for Serpent 2 was started.

- Burnup calculation routines based on the CRAM matrix exponential solver were completely re-written for Serpent 2. The accuracy and convergence of the method were further studied and the results published in Nuclear Science and Engineering. Practical implementation of CRAM is the topic of a conference paper accepted in PHYSOR-2012.

- Advanced predictor-corrector algorithms with higher-order approximations and sub-stepping were implemented in burnup calculation. The work was carried out in collaboration with the NEPAL project, and example results will be presented in PHYSOR-2012.

- Development of a calculation routine capable of handling the temperature dependence of reaction rates during the transport simulation was started. The methodology is described in a paper accepted for publication in Nuclear Science and Engineering, and some first results will be presented in PHYSOR-2012. The capability to model temperature distributions at high accuracy is new to Monte Carlo calculation, and the routine forms the basis for the simulation of temperature feedbacks and multi-physics coupling. The method is already used in a fuel temperature feedback model, developed as an M.Sc. project at Aalto University.

- International collaboration and interaction with the Serpent user community included distributing two updates to Serpent 1, participating in an international reactor physics conference M&C 2011, maintaining a website and a discussion forum¹ and direct communication with users via e-mail. The First International Serpent User Group Meeting was organized with Helmholtz-Zentrum Dresden-Rossendorf (HZDR) in Dresden, Germany, September 15-16, 2011. The two-day event brought together 33 Serpent users from 15 organizations around the world.

1 Serpent website: http://montecarlo.vtt.fi Serpent discussion forum: http://ttuki.vtt.fi/serpent
Recent publications include 3 scientific review articles, 5 conference papers and 1 research report.

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.

**Specific goals in 2011**

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. 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.

On the basis of obtained results, the exact internal temperature distribution of a fuel pellet is modeled. For this purpose, methods are developed for temperature-dependent neutronics calculation.

We apply basic physics in various size scales in these calculations. We aim at increasing understanding of the physical background of macroscopic phenomena. Where applicable our results will be compared to those obtained with traditional computer codes.

![Figure 2.3.4.1. Relative errors in the atomic densities of $^{99}$Mo and $^{131}$I in the PWR and SBU assembly test case with various numbers of substeps on the corrector. The PWR calculations used 22 steps while the SBU ones used 10.](image-url)
Figure 2.3.4.2. Deviation of the continuous temperature profile of the Gd-doped VVER-pin from a parabolic temperature distribution at different burnups (%) (heat productions and material compositions obtained from the multi-temperature calculations).

Deliverables in 2011

- The most important result of the project is education of new experts on nuclear engineering and the sustainable recruiting of new students to an academic research and education environment. We expect the first doctoral theses to be ready by the end of SAFIR2014 programme. Additionally, a few master's theses are made annually in the research group. Teaching of nuclear engineering and recruiting of new experts is done in close collaboration with VTT, STUK and the power companies. It is essential that the group maintains a critical mass before the new experts leak to project work at other organizations. In 2011, three graduate students and one undergraduate student contributed to NEPAL.

- A group of methods for burnup calculations solves the changes in material compositions by evaluating an explicit solution to the Bateman equations with constant microscopic reaction rates. This requires predicting representative averages for the one-group cross-sections and flux during each step, which is usually done using zeroth and first order predictions for their time development in a predictor–corrector calculation.

In one of our papers we present the results of using linear, rather than constant, extrapolation on the predictor and quadratic, rather than linear, interpolation on the corrector. Both of these are done by using data from the previous step, and thus do not affect the stepwise running time. The methods were tested by implementing them into the
reactor physics code Serpent and comparing the results from four test cases to accurate reference results obtained with very short steps. Linear extrapolation greatly improved results for thermal spectra and should be preferred over the constant one currently used in all Bateman solution based burnup calculations. The effects of using quadratic interpolation on the corrector were, on the other hand, predominantly negative, although not enough so to conclusively decide between the linear and quadratic variants.

- When material changes in burnup calculations are solved by evaluating an explicit solution to the Bateman equations with constant microscopic reaction rates, one has to first predict the development of the reaction rates during the step and then further approximate these predictions with their averages in the depletion calculation. Representing the continuously changing reaction rates with their averages results in some error regardless of how accurately their development was predicted. Since neutronics solutions tend to be computationally expensive, steps in typical calculations are long and the resulting discretization errors significant.

In one of our papers we present a simple solution to reducing these errors: the depletion steps are divided to substeps that are solved sequentially, allowing finer discretization of the reaction rates without additional neutronics solutions. This greatly reduces the discretization errors and, at least when combined with Monte Carlo neutronics, causes only minor slowdown as neutronics dominates the total running time.

Substeps, as well as the higher predictor and corrector orders, will be implemented to the release branch of Serpent starting from version 2.0, which is expected to be released in 2012. The results are, however, not Serpent specific and could be applied to other codes using similar methods.

- We derived a formalism to order nuclides by their importance to a given quantity of interest. A quantity can be any result of any linear effect of nuclides. Linear effects yield quantities that depend linearly on the amounts of nuclides, such as decay heat emission rate, spontaneous fission rate and neutron balance rate. Non-linear effects yield quantities such as effective multiplication factor and detectability of nuclides, and depend non-linearly on the amounts of nuclides.

The formalism can be used in identification of important minority and majority nuclides for any application. Application-specific criteria must be expressed as quantities of interest for which effects must be derived. This part remains to be an art. The formalism can be used to order nuclides by their importance to selected quantities of interest. Another art is to judge which nuclide is the last important one for a quantity; any nuclide with less importance is not important.

- We developed an external temperature solver for a fuel pellet, named Thermoss, and coupled it to Serpent in the burnup calculations. For the burnup calculations the fuel pins were divided into rings. Material compositions and heat production in each ring were obtained from Serpent. They were utilized to first calculate the local heat conductivity and then solve the steady-state radial heat conduction equation analytically in each ring. The outer temperature of the cladding was used as a boundary condition. Since local heat conductivity is dependent on the solved temperature distribution, iterations were made in calculating the heat conductivity and the accurate temperature distribution until self-consistence was obtained.
Using this code, we estimated the errors that result from using a homogenized temperature for gadolinium doped fuel pins rather than a more realistic multi-temperature profile. The first case focused on an infinite square lattice of EPR pins, where a Gd-doped pin was surrounded by eight non-doped pins. Comparisons were done at burnups ranging from 0 to 60 MWd/kgU, showing different trends for different temperature homogenization methods. The most accurate results were obtained from Rowlands’ chord-averaging that was also the only method that stayed on the conservative side, i.e., overpredicting the reactivity rather than underpredicting it.

The second case was set to assess the errors made in assuming a bundle-wide homogeneous temperature in a Lovisa-type VVER-bundle containing six Gd-doped pins. Differences began to show at burnups of 20 MWd/kgU and higher. The results of the VVER-case state that even though the volume average temperature of the gadolinium doped pins reaches the homogenized temperature at a certain burnup, the difference between the cases only continues growing at an accelerated pace. One cannot thus justify using a homogenized temperature on the basis that the average temperature of the Gd-pins will reach it eventually.

We concluded that while the effect of temperature homogenization on the neutronics of the system is very subtle, it results in clear differences in the burnup calculations. The differences are explained by changes in the material compositions due to the temperature related changes in cross-sections.

- We started to construct a mesoscopic simulation model of the thermal creep failure of fuel pellets. The model will eventually include 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 this model continues in 2012.

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 2011

Specific goals for 2011 include initialization of the validation project for the fuel behaviour code ENIGMA, update of the statistical versions of the steady state fuel behaviour codes, improvement of selected models and a study supporting a utilization of the statistical LOCA tool developed during SAFIR2010 project POKEVA. Also a strong interdisciplinary effort was
planned to better utilize the fuel behaviour studies, including a review mapping the consequences of the rod failure with the focus being eventual bridging between the domains of fuel behaviour and severe accident studies, a creation of a coupling between a fuel code and a reactor physics code and a study to determine the minimum parameters that can be used to characterize a fuel rod usable in applications where the exact power history is not known. 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 2011**

- A review on processes relevant to failed fuel rods was made. Both LOCA and RIA conditions and consequences were discussed, and insights into radionuclide release were provided.

- Serpent and ENIGMA were coupled by a code SEN capable of using temperature and dimensional data from the fuel code in the neutronics code and power deposition information from Serpent in the fuel behavior code, thus enabling new kind of investigation into the assumptions that are customarily made for the lack of accurate data.

![Relative differences in homogenized cross sections, Serpent-SEN](image)

*Figure 2.3.5.1. The relative difference in homogenized cross sections when using the thermomechanical data from ENIGMA in a Serpent burnup calculation as compared to the traditional approach.*

- A statistical Perl routine was created for ENIGMA, enabling easier upkeep of the code in the future. A study on the characterization of a VVER rod was performed, enabling future creation of the simplified fuel rod model.

- A study on the global parameters affecting the variation of system codes providing LOCA boundary conditions to the fuel behaviour codes was performed. New boundary conditions were implemented into the FRAPTRAN-GENFLO coupled fuel behaviour / thermal hydraulics code for improved simulation of LOCA conditions.

- A computer code SPACE (Simulation Performance Analysis Code for ENIGMA) for simulation validation work was created and a database containing cases based on IFPE database was initialized.
Figure 2.3.5.2. Gas gap width as a function of burnup as described by ENIGMA calculations and the corresponding simplified model.

- Halden IFA-650 LOCA experiments 2, 3 and 4 were analysed in co-operation with Quantum Technologies (Sweden).

- FUMEX-III CRP participation begun as a part of POKEVA in 2009 was successfully finished and the simulated cases reported to IAEA. A Licentiate thesis was done based on the work done to the benchmark programme.

- WGFS RIA benchmark calculations were performed with SCANAIR code and the results reported in the benchmark conference. As a part of on-going SCANAIR co-operation with IRSN a researcher exchange was successfully initiated.

- ENIGMA cladding creep correlations were revised, and calculations of Halden IFA-610 overpressure series were redone. Both were presented at the Enlarged Halden Programme Group 2011 meeting. Further on, the PIE results reported for the IFA-610 series were collected and reviewed.

- Several international conferences were participated in. Information event on international fuel behaviour projects was organized in June 21st.

2.4 Thermal hydraulics

In 2011 the research area "Thermal Hydraulics" consisted of eight projects: Application of best estimate plus uncertainty evaluation method (BEPUE), 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) and Modelling of pressure transients in steam generators (SGEN).

2.4.1 Application of best estimate plus uncertainty evaluation method (BEPUE)

The objectives of the project is to choose the most important parameters, which affect the results of a large break loss of coolant accident (LBLOCA) analysis and to define the suitable
variation range for those parameters. Finally the chosen parameters will be applied to an exemplifying best estimate plus uncertainty analysis of LBLOCA.

**Specific goals in 2011**

1. Identification of input parameters
The main emphasis in the project will be in choosing the input parameters for uncertainty evaluation and defining the appropriate uncertainty range for those parameters. Part of the work will be done by the reactor dynamics team in VTT (SAFIR research area 3). The most important general parameters affecting core coolability in case of LBLOCA will be defined. The report of OECD/BEMUSE will be used as a starting point. The special features of the new power plant concepts will be taken in account as far as possible.

2. Application to LBLOCA
As an example of the BEPU method an application to LBLOCA in a typical modern PWR plant will be calculated. An APROS input for a generic PWR plant will be prepared and a procedure for the statistical analysis will be defined.

**Summary of results**

The SUSA (Software für Unsicherheits- und Sensitivitätsanalysen) method developed by the GRS was used in the project. In the method, a set of parameters of uncertain accuracy is selected. A set of initial values is drawn randomly from the set of parameters, and the same transient is calculated with each set. The sample obtained as a result can be analysed with statistical methods. The most important parameter for variation was chosen based on the OECD/BEMUSE project results and on literature review. The method does not restrict the number of the parameters; the properties of the used code and model and more likely to be the restricting factors. Some parameters, which are recommended in OECD's BEMUSE programme, but which could not be varied in a natural manner in the current APROS Testing Station environment, were identified in the project.

As the purpose of the BEPUE project was not to calculate a real accident analysis, but to study the uncertainty method and its application in APROS environment, the plant model used did not need to be an accurate and well validated. In order to implement the educational objective of the research programme, it was decided to construct an approximate model of APR1400, a Korean pressurised water reactor, which features two vertical steam generators, two cold legs per steam generator and flow restrictors in emergency accumulators (Figure 2.4.1.1). All plant data used in modelling have been collected from public sources. For this reason, the model is not accurate in all respect and the results are merely approximate. However, the model can be used as a basis for subsequent modelling needs.

Large break loss of coolant accident (LBLOCA) was calculated as an application of the uncertainty analysis. The analysis was carried out using the uncertainty tool in the APROS Testing Station. For the selected parameters 93 sets of initial values were drawn randomly and the same pipe break accident was calculated for each case. The main output parameter of LBLOCA, the peak cladding temperature (PCT), is presented in Figure 2.4.1.2.

The APROS Testing Station can be used also to carry out a sensitivity analysis in order to identify the most significant parameters. The tool calculates the Spearman correlation coefficient (RCC) and the Kendall τ coefficient. The sensitivity analysis was carried out on the PCT. The discharge coefficient of the critical flow and the containment pressure proved to be the most significant parameters.
Figure 2.4.1.1. APROS model of the APR-1400.

Figure 2.4.1.2. Peak cladding temperature (PCT) in the 93 calculation runs.
Deliverables in 2011

- Method to apply uncertainty evaluation in LBLOCA and reported example of the application.
- An indicative APROS model of the APR-1400 for further applications

2.4.2 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. 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.

The new power plant concepts offered have passive emergency condenser, which operation is based on natural circulation. The ability of APROS code to simulate behaviour of passive condensers was tested by calculating NOKO and PANDA PCC experiments. The German NOKO experiment facility is a model of the passive emergency condenser of the Kerena plant concept (Figure 2.4.2.1).

![Figure 2.4.2.1. Principal diagram of Emergency Condenser and APROS model of NOKO experiment.](image)

Measured and calculated condenser performance was compared in 10 different operation conditions using 3 alternate condensation correlations available in APROS code. The results show reasonable agreement with the experiments although the comparison to the data suggests that there may be some over prediction of heat transfer in some of the simulated
cases with the default correlation (Shah). Therefore Nusselt model for condensation is recommended for this kind of condensers (Figure 2.4.2.2).

![Figure 2.4.2.2. Calculated and measured heat transfer in NOKO experiments.](image)

TRACE and APROS codes were validated with integral experiments like PWR PACTEL benchmark in LUT and OECD/ROSA2 experiment in LSTF test facility in Japan. The LSTF is a large, 1/48 volumetric scaled, model of a PWR (Figure 2.4.2.3). The counter parent experiment with OECD/PKL2 research program studied hot leg SBLOCA. One of the main objectives of the experiments was to study core exit thermocouples (CET) ability to indicate core overheating. Considerable CET delay was seen in an earlier experiment in the OECD/ROSA research programme. TRACE and APROS codes reproduced the experiment quite well, but unfortunately the results cannot be shown here because the date is restricted to the participants of the research programme.

![Figure 2.4.2.3. LSTF test facility and TRACE model of the primary circuit.](image)
ROCOM is a test facility simulating a KONVOI type PWR. The test 2.1 simulates a pseudo-steady-state of test PKL III G 3.1 at a moment, when cold water from the loop with broken steam line flows to downcomer. Verification of APROS and TRACE 3D vessel models was continued with this test, where the cold water plume reaches the lower part of the downcomer. Measured and calculated temperature distributions are illustrated in Figure 2.4.2.4. Geometrical limitations of the models restricted the mesh size to the size of the connected cold leg pipe. Despite of coarse grid the temperature distribution in the downcomer was quite well reproduced. However the model should be further validated before power plant applications.

![Figure 2.4.2.4 APROS and TRACE 3D vessel models (left) and measured and calculated cold water plume in the ROCOM test.](image)

**Containment thermal hydraulics**

A Generic Containment code-to-code comparison benchmark was continued in the frame of SARNET-2 WP7. The second step of the benchmark simulated severe accident conditions where hydrogen and carbon monoxide and dioxide (products of core concrete interactions) were released to containment. VTT participated also in the second step of the benchmark calculation both the blind and open phase with sensitivity studies with the APROS Containment. A rather simple containment model based on a German 1300 MW<sub>e</sub> PWR was developed by FZJ and delivered to participants i.e. all participant used a similar lumped parameter nodalisation to allow detailed code to code comparison (Figure 2.4.2.5).
Figure 2.4.2.5. Control volumes within the reactor building.

In total, calculation results from 14 different organizations applying 9 different codes were submitted. The APROS result of containment pressure and hydrogen concentration in the dome is compared with the mean value and standard deviation band of the other calculations in Figure 2.4.2.6. Similar trend was observed in other target variables compared.

Figure 2.4.2.6. Calculated containment pressure and hydrogen concentration compared to the mean value in SARNET Generic Containment benchmark.

A single droplet heat and mass transfer tests conducted at IRSN CARAIDAS facility (SARNET spray benchmark) was calculated using the APROS containment code and Fluent CFD code. Both evaporation and condensation cases were studied. The main goal of the work was to validate the containment spray heat and mass transfer models of APROS and to verify that the droplet models in Fluent are suitable for modelling containment spray. As an example of calculated cases, evaporation of a droplet is presented as a function distance from the spray nozzle. Sensitivity analysis and different modelling options were studied with both codes (Figure 2.4.2.7).
Figure 2.4.2.7. Calculated droplet size evolutions with APROS and Fluent and measured droplet diameters [ERMSAR 2012, paper 5.1].

Deliverables in 2011

- APROS was validated for analysis of passive condenser with NOKO and PANDA PCC experiments.

- APROS and TRACE codes were validated with the integral experiment of OECD/ROSA2 research program studying hot leg SBLOCA.

- APROS was validated with posttest calculation of PWR PACTEL benchmark

- TRACE and APROS 3D vessel models were evaluated by modeling ROCOM downcomer mixing tests.

- APROS containment was validated with SARNET generic containment severe accident specific second phase with H₂, CO and CO₂ release.

- Coupled APROS containment and APROS thermal hydraulic models were validated with PANDA PCC experiments.

- Fluent model of the PANDA heat exchanger experiment was further refined and models were qualified.

- APROS containment and Fluent models of containment spray were validated with SARNET single droplet evaporation and condensation tests.
2.4.3 Experimental studies on containment phenomena (EXCOP)

The objective of the project is to gather an extensive experimental database on condensation dynamics, heat transfer and structural loads, which can be used for testing and developing computational methods used for nuclear safety analysis, such as Fluent, Star-CD, 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 existing PPOOLEX test facility at Lappeenranta University of Technology (LUT), including models of the drywell and wetwell compartments and withstanding prototypical system pressure, is 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 were installed to the PPOOLEX facility in 2011. The applicability of the measurement results for the validation of CFD, lumped parameter and structural analysis codes will increase considerably. Networking among international research organizations was enhanced via participation in the NORTHNET framework and NKS/ENPOOL project.

Specific goals in 2011

Specific goals in 2011 included two experiment series with the PPOOLEX test facility, designing a Passive Containment Cooling System (PCCS) condenser and simulation of several previous experiments with NEPTUNE_CFD and TransAT codes.

In the first experiment series the objective was to study the effect of a blowdown pipe outlet collar on condensation related loads. The shape and geometry of the blowdown/vent pipe outlet designs are known to have a significant effect on structural loads experienced by submerged condensation pool structures. A scaled collar model was installed to the blowdown pipe outlet of the PPOOLEX facility. Additional pressure sensors were attached to the pipe outlet in order to gain high resolution data of the movements of the condensation front during steam discharge through the collar (Figure 2.4.3.1). A series of reference experiments without the collar was first carried out. Then experiments with the same test conditions but using the collar pipe were done.

A more harmonized pattern in the general trend of the signal measured by the pressure sensors at the blowdown pipe outlet can be observed in the collar experiments than in the reference cases. The movement of the condensation front after a rapid collapse of a steam bubble seems to better follow the shape and surface of the pipe wall when there is a collar at the outlet. However, the diminishing effect of the collar on pressure loads could not be verified on the basis of these experiments.

The overall goal of the experiment series on direct contact condensation (DCC) was to produce CFD grade measurement data of rapid steam condensation processes to be used in the development and validation of simulation tools by VTT and KTH. The PIV measurement system, purchased in the ELAINE infrastructure project, was delivered in October. The system was first extensively tested in a separate facility. After that the system was installed to the PPOOLEX facility. New viewing windows were installed to the wet well pool for the lasers and cameras of the PIV system.

The planned experiments on DCC were delayed due to the late arrival of the PIV measurement system and the time needed for its installation and testing. Therefore the experiments carried out during the testing of the PIV system in the PPOOLEX facility were used as an input for this subtask. The quality and the quantity of measured data from the
PPOOLEX experiments will increase remarkably when the PIV system is in use. The aim is to produce CFD grade measurement data of rapid steam condensation processes to be used in the development and validation of simulation tools. The first application of the system in the commissioning phase could be considered successful and the potential of the system was proved (Figure 2.4.3.2).

Figure 2.4.3.1. Instrumentation of the blowdown pipe in the collar experiments.

In certain (severe) reactor accident scenarios the PCCS condenser removes excess core heat. In addition to cooling it also has an important role in mitigating offsite dose by retention of fission product release in containment. The possibility of using PPOOLEX as a host facility in the PCCS experiments in order to gain experimental data for the validation of severe accident code MELCOR was examined.

In discussions with modellers from VTT the horizontal PCCS design used in the ABWR concept was ranked as the most interesting case to be studied in this subtask. Thermal hydraulic conditions attainable in the PPOOLEX facility were considered suitable for the planned experiments. A scaled PCCS condenser model of the type used in ABWRs was designed.
Figure 2.4.3.2. Averaged velocity field of a strong outflow phase with background image obtained with the help of the PIV system in a PPOOLEX test.

The work with NEPTUNE_CFD and TransAT codes, being developed in the EU/NURISP project, continued with the simulation of selected POOLEX experiments. 2D and 3D simulations of a chugging related experiment STB-28 and 3D simulations of a quasi-steady steam-water interface experiment STB-31 were done.

In the NEPTUNE_CFD simulations of the POOLEX STB-28 experiment using a 2D-axisymmetric geometry the condensation models of Hughes-Duffey and Lakehal et al. 2008 were compared. Initial 2D results indicate that the Hughes-Duffey condensation model predicts clearly higher condensation rates than the model of Lakehal et al. 2008. Furthermore, both models indicate lower condensation rates than observed in the experiment. When the steam/water interface was initialized to high enough level inside the pipe, the vigorous initial penetration into the pool water created a turbulent wake. This invoked the chugging due to the improved heat transfer rate by increased turbulence production. The initiated chugging state was then self-sustaining. In that case the condensation models of Hughes-Duffey and Coste-Laviéville predicted realistic chugging behaviour whereas the model of Lakehal 2008b still under-predicted the condensation rates.

**Deliverables in 2011**

- A series steam discharge experiments, studying the effect of a blowdown pipe outlet collar on condensation related loads, was carried out. Additional instrumentation at the pipe outlet and a high speed camera pointing through the pool bottom window into the blowdown pipe helped to track the movement of the condensation front after the collapse of steam bubbles. Two different pool water temperature levels and two different steam flow rates (belonging to the chugging region of the condensation mode map) were used. Reference experiments with the same test conditions were done without the collar. The experiments demonstrated that the collar design has an effect on the rapid condensation phenomenon at the blowdown pipe outlet. The general trend of the formation and collapse of steam bubbles was less chaotic with the collar. However, the dynamic loadings experienced by submerged pool structures were not lower with the collar in these experiments.
The first application of the new PIV measuring system with the PPOOLEX facility as a part of the commissioning phase was successful. The PIV system was also tested in conditions, where the DCC phenomenon was prevailing and the potential of the system was proved.

The possibility of using PPOOLEX as a host facility in the PCCS experiments in order to gain experimental data for the validation of severe accident code MELCOR was examined and a scaled PCCS condenser model of the type used in ABWRs was designed.

POOLEX experiments STB-28 and STB-31 were simulated with NEPTUNE_CFD and TransAT codes. In the STB-31 simulations the Hughes-Duffey condensation model over-predicted the condensation rate by one order of magnitude whereas the Lakehal et al. (2008b) condensation model predicted the condensation rate very accurately. The Coste-Lavíevelle model predicted condensation rates close to the rates of the Lakehal et al. condensation model.

NORTHNET Roadmap 3 meetings were participated in Lappeenranta and Västerås. Status of condensation pool research in participating organisations (LUT, VTT and KTH) was presented and Roadmap 3 plans were updated for the coming years. A combined ENPOOL-NKS funding application for 2011 by LUT, VTT and KTH was written.

2.4.4 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 and methods are limited. In addition, the license policy prevents cost effective utilization of parallel computer resources. Better utilization of parallel computing would allow the use of more complex physical modeling and detailed computational grids. This would improve the accuracy and reliability of analysis and increase the number of situations where CFD methods can be utilized.

The main aim of the project is to validate the open source CFD code OpenFOAM as a tool for nuclear safety related simulations. The Finnish OpenFOAM community in the field of nuclear safety is formed. In single-phase flow simulations, the aim is to simulate flow and heat transfer in applications such as T-junction of pipes and fuel rod bundles by using accurate computational models and to effectively utilize parallel computing. As a result a more detailed understanding of coolant mixing is achieved.

An OpenFOAM based Euler-Euler two-phase solver is developed and enhanced for nuclear safety applications. The solver is modified for calculations of water-air and water-steam mixtures, two-phase heat transfer and subcooled nucleate boiling. The longer term goal is to be able to simulate subcooled nucleate boiling in fuel rod bundles. The code is validated against experimental data obtained from literature and simulations with other codes. The same solver framework is also to be utilized in simulation of direct-contact condensation, once suitable models have been implemented.

Specific goals in 2011

The flow behavior and heat transfer in VVER-440 fuel rod bundles was intended to be studied using RANS and DES turbulence models, but effective utilization of DES proved too time consuming at this stage and was abandoned. However, the behavior near the spacer grids was
studied with separate detailed DES simulations. Figure 2.4.4.1a shows a detail of velocity and turbulent kinetic energy in a VVER-440-simulation. The behavior of DES models was studied further in the same OECD/NEA T-junction benchmark that LES simulation results were submitted to in 2010. In the benchmark, thermal mixing was simulated in a T-junction of pipes in an experiment performed at Vattenfall R&D.

An international OECD/NEA benchmark, MATiS-H (Measurement and Analysis of Turbulent Mixing in Subchannels - Horizontal), is participated. In the benchmark, turbulent flow in a rectangular fuel bundle is solved using OpenFOAM. Different turbulence models are tested. The case is also simulated with Fluent so that comparisons can be made with OpenFOAM. The work started in 2011 with mesh building, mesh sensitivity studies and preliminary calculations with one subchannel. Final results and a report will be delivered by the benchmark deadline in end of April 2012. Figure 2.4.4.1b shows the velocity profile in one plane in the Matis-H model.

The two-phase solver introduced in 2010 has been develop further by introducing heat transfer, non-uniform material properties and models for interfacial boiling and condensation, as well as an implementation of the so-called RPI wall boiling model. In addition, the solution algorithm has been refined and turbulence modeling corrected to match the single phase results at the dilute limit. The solver was validated in two sub-cooled nucleate boiling DEBORA experiments carried out by CEA. The results were compared to experimental results and numerical results obtained with commercial code Fluent.

The simulations were carried out in parallel and parallel scaling of the solver was tested up to 48 threads with promising results. Overall the results of the OpenFOAM based solver compared well both to the experimental results and numerical results obtained with Fluent, but the turbulence model proved highly sensitive to vapor volume fraction in the near wall cells. This caused an under prediction of turbulent mixing. Both the OpenFOAM based solver and Fluent had small errors in the energy balance that require further investigation.

The direct condensation simulations were prepared for by testing a development version of the two-phase solver in an adiabatic test case based on POOLEX STB-31 experiment. The simulations were carried out successfully, but further investigation is needed regarding inter-phase drag and turbulence modeling.

Figure 2.4.4.1. a. Velocity [m/s] and turbulent kinetic energy [m$^2$/s] contours in a VVER-440 spacer grid simulation. b. Velocity profile [m/s] in flow through the Matis-H fuel bundle. Spacer grids and mixing vanes are visible in grey (a & b).
Figure 2.4.4.2. Contours of simulated vapor volume fraction and liquid temperature (left) as well as comparison of simulated and experimental radial volume fraction profile at the end of the heated section (right).

**Deliverables in 2011**

- Reports concerning single-phase flows: Two reports on simulation of turbulent mixing in VVER-440 fuel rod bundle and a report on DES simulation of the OECD/NEA T-junction benchmark experiment.

- Reports concerning two-phase flows: Documentation of two-phase solvers: an intermediate 0.2 version and the current 0.3 version. Reports on the DEBORA subcooled nucleate boiling validation simulations and the preliminary adiabatic test simulations of POOLEX experiments.

- An internet portal to distribute reports and code within the project.

### 2.4.5 Numerical modelling of condensation pool (NUMPOOL)

Two-phase and Fluid-Structure Interaction (FSI) calculation methods are developed that can be used for modelling of a pressure suppression pool of a boiling water reactor (BWR). In particular, the direct-contact condensation occurring in the pool during a hypothetical large-break loss-of-coolant accident 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 in the EXCOP project.

The NUMPOOL project is part of 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.
Specific Goals in 2011

In 2011, calculations of direct-contact condensation in the pressure suppression pool have been performed. Partial pressure model for the condensation of pure vapour is now used instead of diffusion model that was previously used for the condensation in a mixture of vapour and air. The partial pressure model makes possible modelling of the condensation of pure vapour without any non-condensable gas. The CFD mesh has been refined in order to reduce numerical diffusion near the vent outlets. Numerical diffusion is also reduced by the use of second order time discretization and second order spatial discretization for all variables. Turbulence has so far been modelled with the standard \( k-\varepsilon \) model of FLUENT for the mixture of phases, which produces very large effective viscosity. The RNG \( k-\varepsilon \) model has been tested for modelling turbulence because it produces smaller effective viscosity.

The chugging phase of the PAR-10 experiment with two vent pipes has been recalculated several times during year 2011 with different versions of the direct-contact condensation model. In the last simulations, somewhat improved results for chugging were obtained. Clear differences compared to the experiment still exist, for instance, in the chugging period. In Figures 2.4.5.1 and 2.4.5.2, condensation of vapour in two vent pipes submerged in cold water is illustrated at one instant of time. Note that the scale of the condensation rate is logarithmic.

The rapid collapse of a large steam bubble in PPOOLEX experiment COL-01 has been analysed with the new Eulerian model of ABAQUS. The calculated pressure peak became clearly too high and narrow, even though the collapse time of the vapour bubble was correct. The low and wide experimental pressure peaks may be caused, e.g., by breaking of the bubble into two-phase mixture or rise of steam pressure in the late phase of the collapse. By observing the collapse behaviour, the pressure variation inside the bubble was fitted with the experiment. The pressure variation can be used for estimating the condensation rate of steam in future. The effect of system size on the pressure peak was also examined; these results can be used for studying more thoroughly the scaling of the experimental results to full-scale in future.

The desynchronization of chugging events in the two vent experiment PAR-10 was studied. The statistical distribution of desynchronization was determined from the measured pressure data and compared to results obtained in a seven vent pipe experiment found from literature. The distribution of desynchronization times in the PAR-10 experiment is shown in Figure 2.4.5.3. The standard deviation of the desynchronization times was found to be 38 ms in this experiment.

The response of BWR containment during desynchronized chugging events and with varying speeds of sound was numerically computed using direct time integration and modal dynamics procedure available in ABAQUS. The preliminary results show significant decrease in the loads experienced by the containment, when the desynchronization is taken into account. In Figure 2.4.5.4, the calculated von Mises stresses at the pool boundary are illustrated for a unit load. The values of the stresses for real loads are obtained by linear scaling of the result.

Deliverables

- CFD simulation of direct-contact condensation of pure vapour in a PPOOLEX experiment
- FSI calculation of a PPOOLEX experiment with a large condensing steam bubble
- FSI calculation of blowdown with a sector model of a BWR
• Report on the CFD and FEM calculations.

Figure 2.4.5.1. Volume fraction of vapour in two vent pipes submerged in a water pool.

Figure 2.4.5.2. Condensation rate (kg/m$^3$/s) in two vent pipes submerged in water pool.

Figure 2.4.5.3. Histogram of the desynchronization time of the peak underpressure caused by collapsing vapour bubbles in different vent pipes.

Figure 2.4.5.4. Vertical cross-section of a BWR containment at the time of occurrence of maximum von Mises stresses at the pool boundary. The stresses have been calculated for a unit load.
2.4.6 Improvement of PACTEL facility simulation environment (PACSIM)

The main objective of the PACSIM project is to enhance the utilization of the TRACE thermal hydraulic code and to improve the simulation environment of the PACTEL facility. This project is a straight continuum to the earlier PACSIM project, but its focus in 2011 is only on VVER modelling. PACSIM was concentrated on calculations of primary to secondary leakage (PSL) experiments. The project also summarizes the knowledge that has been acquired of the TRACE code modeling of VVER PACTEL facility so far. Option for continuing the project during 2012 has been preserved. The planned calculation cases are Anticipated Transient without Scram (ATWS) experiments.

Specific goals in 2011

Specific goals in 2011 included calculations of primary to secondary leakage (PSL) experiments. The chosen experiments were safety critical cases since the main goal was to find the worst possible conditions during which the leak flow reversal can occur. In Loviisa Nuclear Power Plant, the construction of the steam generator primary collectors has been changed. If the steam generator collector rupture occurs, the leak flow area is significantly less than in old construction. The maximum possible flow area corresponds now to a rupture of five to six heat exchange tubes. In experiments PSL-10 and PSL-11 these construction changes were investigated. In the experiment PSL-10 the original construction was used as a reference for the second experiment. The experiment PSL-11 focused on effect of the new construction. The experiment procedures were based on the current regulations for operator actions during a state of emergency in the Loviisa nuclear power plant.

In each experiment all three steam generators were used, two intact and one broken steam generator. The pressurizer spray and the accumulators were used in all the experiments. Due to different collector construction in PACTEL, the uppermost heat exchanging tube row was chosen as a break location. This break setup was also used for the calculation model.

The experiments PSL-10 and PSL-11 were then calculated using the TRACE code model (Figure 2.4.6.1). The break diameters were 5.5 and 2.5 mm. Some model changes and improvements were introduced concerning steam generator behavior. For example, it was necessary to restrict the inner circulation of the horizontal steam generator to model the pressure behavior more precisely. In experiment PSL-10 the valve operations concerning the broken steam generator had significant effect on the overall behavior of the calculations. In calculation of PSL-11 the primary side pressure was not decreased as expected. Some improvement could be implied by modifying the break loss coefficient.
Figure 2.4.6.1. Primary and secondary side pressures in experiment PSL-10 vs. TRACE calculation.

Deliverables in 2011

- PACTEL experiments PSL-10 and PSL-11 were calculated with the TRACE model. These experiments described the primary to secondary side leakage phenomenon, which contain very important safety critical issues. The TRACE calculation results pointed out that some model modifications were still necessary, especially concerning modeling of horizontal steam generators. Effects of certain parameters were investigated with some sensitivity studies. With these studies the calculation results could be improved and the overall results agreed better with the experiment results. The validation report described the results in more detail.

- Conference paper titled: “Modeling of the PACTEL Facility and Simulation of a Small Break LOCA Experiment with the TRACE V5.0 Code” was prepared for the 14th International Topical Meeting on Nuclear Reactor Thermalhydraulics, NURETH-14, which was held in Toronto, Ontario, Canada, on September 25-30, 2011. The paper presented the TRACE calculation results of earlier small break LOCA experiments with PACTEL facility.
Figure 2.4.6.2. TRACE/SNAP model of PACTEL (only one loop presented) used for calculations of primary-to-secondary-leakage experiments.

2.4.7 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. Participation in OECD projects with PWR PACTEL was negotiated in connection with the OECD PKL3 project. The possibilities to arrange an OECD International Standard Problem (ISP) on PWR PACTEL was also studied.

The PWR PACTEL test program continued with two test series, natural circulation in low pressure and flow reversal in steam generator tubes. Also possibilities of measuring the void fraction in the secondary side of vertical steam generators were studied. Possibilities of using different measuring and instrumentation systems as well as using separate effects test facilities were studied to find a solution for this problem.
Specific goals in 2011

Specific goals in 2011 included PWR PACTEL benchmark, PWR PACTEL tests, preparation of international cooperation, and solving the void measuring problem in the secondary side of the vertical steam generators. One of the first PWR PACTEL experiments was chosen to be a benchmark calculation test transient to offer internationally a possibility to check the analysis skills of the analysts in organizations performing thermal hydraulic safety analyses of nuclear power plants. PWR PACTEL benchmark was launched in a workshop in October 2010 in Lappeenranta. The second workshop was arranged on 12.-13.5.2011. Seven organizations from Czech Republic, Germany, Italy, Sweden, and Finland participated to the benchmark exercise, and they used four different system codes within the exercise case.

A series of experiments on natural circulation in low pressure and related computer analyses were performed to support the earlier characterizing experiments in higher pressure. The goal of the test series was to confirm low pressure behavior observed in the ROSA facility also in the PWR PACTEL facility. These phenomena (the system wide oscillation for the mass inventories between 40 % and 50 %, a characteristic C-shape vertical temperature distribution in the steam generator secondary side, and a “stagnant two-phase stratification” at low pressures) were not found in the PWR PACTEL experiments.

![Figure 2.4.7.1. Relative loop flow rate as a function of the primary mass inventory in the single loop SIR-32 and SIR-33 experiments with 115 kW core power. Also the results from the high pressure experiment SIR-31 are presented as a reference.](image)

The preferential flow reversal in the long U-tubes has been observed in several experiments performed with the LOBI, SEMISCALE, BETHSY, PKL, and ROSA/LSTF facilities. Such a flow behavior strongly affects the heat transfer efficiency of a UTSG. The second series of PWR PACTEL experiments was carried out to experimentally verify the flow reversal in the steam generator tubes. From the measured data was estimated that the flow reversed in about 30 % of the heat exchange tubes. According to these experiments the amount of tubes where flow was reversed is independent of the secondary side conditions. The location of the hot leg connection seems to have an effect on the flow reversing in the heat exchange tubes.

Possibilities to measure void fraction in the secondary side of the vertical steam generator of the PWR PACTEL nuclear power plant test facility were studied. The aim was to find as cost-
effective and functional way to measure void fraction as possible. Physics of two-phase flow and void fraction were examined and different methods to measure void fraction were introduced. Suitability of each method to the PWR PACTEL test facility was evaluated. Measuring void fraction in the secondary side proves to be very difficult. Therefore, conditions on measuring void fraction in a separate steam generator test facility can be considered.

**Deliverables in 2011**

- The PWR PACTEL benchmark exercise was organized in Lappeenranta, Finland by Lappeenranta University of Technology. The first benchmark workshop, where the blind calculation part was introduced, was held 5th of October 2010. Seven organizations from Czech Republic, Germany, Italy, Sweden, and Finland participated to the benchmark utilizing four system codes. The second workshop, where the calculation results of the blind and open calculations as well as related modeling issues were discussed, was held 12th and 13th of May 2011 in Lappeenranta.

- A series of PWR PACTEL experiments on natural circulation in low pressure and related computer analyses were performed to support the earlier characterizing experiments in higher pressure.

- A series of PWR PACTEL experiments was carried out to experimentally verify the flow reversal in the steam generator tubes.

- PWR PACTEL experiments have been proposed and accepted as support to the PKL experiments in the OECD/PKL3 project.

- Physics of two phase flow has been studied. Void measurement principles and devices have been reviewed and there suitability to the PWR PACTEL steam generators has been estimated. Also the costs have been estimated.

- Measurement results of the experiments in 2011 can be found from the experiment database maintained by the research group.

### 2.4.8 Modelling of pressure transients in steam generators (SGEN)

The objective of the project is to develop a simulation methodology and tool for the modeling of horizontal and vertical steam generators of NPPs, where the multidimensional effects and the two-phase flow phenomena are taken into account. The model developed in the project includes the essential physical phenomena occurring in steam generators, such as heat transfer from the primary to the secondary side and the pressure loss of the two-phase flow in the tube banks of the secondary side.

Three-dimensional simplified model is developed for the calculation of rapid transients in the PWR PACTEL steam generator. The model of the secondary side is implemented in the commercial FLUENT CFD code by using User-Defined Functions. The primary circuit is modeled with APROS. In the CFD model of the secondary side, the geometry of the primary tubes is not described in detail. Instead, a so-called porous media model is used for the pressure loss and heat transfer on the secondary side.
Specific goals in 2011

In 2011, the phase change models are reformulated for the modeling of rapid transients in steam generators. The bulk evaporation and condensation models are formulated with the aid of an interface temperature between the phases. The heat transfer from liquid and vapor phases to the interface is resolved and the heat balance at the interface is calculated. Then, the mass transfer, i.e., evaporation or condensation, is calculated from the heat balance. The evaporation caused by the heat from the primary tubes is treated in the same manner. Note that similar modeling techniques can also be applied to rapid phase changes in other devices that are of interest in nuclear safety analysis.

An interface to the steam tables of APROS is implemented in the steam generator model. Variable vapor properties are described with the user-defined real gas model of FLUENT. The material properties of vapor are first pre-calculated and tabulated from the APROS steam tables for various values of temperature and pressure, and then interpolated from these tables during the FLUENT calculation. Note that the present implementation of steam tables can also be used in other problems in nuclear safety analysis.

Two transient scenarios were selected to test the performance of the steam generator model: a pressure reduction, which could be the result of a small steam line break or a valve malfunction, and a pressure rise, such as the one resulting from a turbine trip. For the purpose of comparing the simulation results to experimental data, two such pressure transients were performed by Vesa Riikonen in the PAX project, in conjunction with experiments RF-02 and RF-04, with the PWR PACTEL facility at the Lappeenranta University of Technology.

Pressure drop and pressure rise on the secondary side

The scenario involving a reduction in the secondary pressure starts from an initial stationary state with a secondary pressure of 4.0 MPa and a power level of about 60 kW. Then, at time $t = 1700$ s, pressure is decreased to 3.0 MPa in approximately 160 seconds.

In Figure 2.4.8.1, the rapid increase of vapor fraction during the pressure drop on the secondary side is shown on a vertical symmetry plane of the steam generator. Vapor fraction increases mainly on the hot side and remains very low on the cold side.

Some discrepancies between the measured and calculated temperatures were found on the cold side of the riser. This suggests that the experimental facility is not fully tight but leaks occur from the hot side of the ring space to the cold side. Therefore, the leaks should be added in the simulation model.

The CFD model was found to predict stronger transients than those observed in the experiments. In the pressure drop transient, the calculated heat transfer from the primary tubes to the secondary side was larger than in the experiment. The production of steam in the transient was, however, dominated by the bulk evaporation caused by the pressure drop. Only a fairly small amount of vapor was produced by the heat transfer from the primary circuit. In the pressure rise transient, the calculated heat transfer from the primary side to the secondary side was smaller than in the experiment.

The one-way coupling of the APROS model of the primary side to the CFD model of the secondary side is one probable reason for the differences between the calculations and the experiments. In addition, the heat transfer correlations used in the CFD model can also be a
source of discrepancies. In future work, the reasons for these differences should be investigated.

![Graph showing volume fraction of vapor over time](image)

**Figure 2.4.8.1.** Calculated volume fraction of vapor [-] on the center plane of the PWR PACTEL steam generator (hot side on the left and cold side on the right).

### Deliverables in 2011

1. Interface between the FLUENT CFD solver and APROS steam tables
2. Report on the validation calculations of pressure transients in a steam generator

**Acknowledgement:** The authors are grateful to Mr. Vesa Riikonen, Ms. Virpi Kouhia and the PAX project for the data of the PWR PACTEL experiments RF-02 and RF-04.

### 2.5 Severe accidents

In 2011 the research area "Severe Accidents" consisted of four projects: Core debris coolability (COOLOCE), Chemistry of fission products (FISKE), Thermal hydraulics of severe accidents (TERMOSAN) and Transport and chemistry of fission products (TRAFI).

#### 2.5.1 Core debris coolability (COOLOCE)

The project aimed at complementing the COOLOCE experiments that investigate the coolability of porous core debris beds and continuing the validation of the simulation codes...
used to model this phenomenon against the experiments. Experimental data of the coolability (i.e. dryout behaviour) of a conical (heap-like) particle bed is compared to that of an evenly-distributed cylindrical bed in order to find out the effect of the bed geometry on the dryout power. The goals of the simulation work are to evaluate the capabilities of the codes to predict the dryout behaviour in power plant severe accident scenarios, and to improve the modelling approaches based on the simulation results.

**Specific goals in 2011**

The main goal in 2011 was performing experiments consisting of series of experiments with the conical and cylindrical debris beds. As a result, comparison data for a range of pressures of the dryout behaviour of the two debris bed geometries was to be obtained. The design of the final specifications of the cylindrical test bed was included in the plan, as well as some modifications to the experimental set-up in order to make the heating arrangement of the test facility more robust to resist high temperatures and the rather challenging conditions within the test beds.

The objective of the analytical work was to perform numerical analysis of the COOLOCE experiments using the MEWA and PORFLO codes. The MEWA 2D code has been developed at the IKE institute at Stuttgart University specifically for investigating debris coolability issues. PORFLO is a multi-purpose in-house code developed at VTT for solving two-phase flow problems encountered in nuclear power plant safety assessment. The MEWA calculations were more directly oriented at post-test code validation while the PORFLO calculations served as demonstrations of the dryout behaviour in 3D. In the long-term, the 3D approach provides possibilities to extend the scope of the simulations to several topics of importance, e.g. highly irregular (more realistic) debris bed configurations and the modelling of the pool region in which the debris bed is immersed.

**Deliverables in 2011**

- Two series of experiments were carried out that consisted of a total of 5 test days. The pressure range in the experiments for which dryout was measured was 1 – 3 bar for the conical test bed and 1 – 7 bar for the cylindrical test bed. It was concluded that the coolability of the conical test bed is approximately 50% poorer for the conical debris bed in case the two debris beds are distributed over an equal area, and presuming that the formation of the first dry zone is the criterion for coolability. This is because of the greater height of the conical configuration and the increased thermal loading in the upper parts of the cone.

- Research reports describing the experiments were written. The technical reports were used as a starting point for a journal article that describes and summarizes the results and findings of the COOLOCE experiments in 2010-2011. The draft of the proposed journal article was submitted to Nuclear Engineering and Design.

- Post-test calculations of the experiments were performed by using the MEWA severe accident analysis code. It was found that the code is capable of predicting the dryout power in the debris bed with a good accuracy. The relative coolability of the conical and cylindrical debris beds was predicted by the code to be close to the one shown in the experiments. A homogenously heated, uniform debris bed model was assumed in the simulations which is the “traditional” assumption used in power plant scenarios.
The PORFLO model of the debris bed was updated by generating a refined mesh model which can be used for e.g. estimating the effect of the heating arrangement and other inhomogeneities. A better approximation of the debris bed surface was also achieved with the refined grid model. Simulations were run with the refined model for the cases of homogenous and non-homogenous (local heating and variable porosity) debris beds, taking into account both conical and cylindrical geometries. In general, the dryout process was reasonably captured in the simulations. However, targets of code development and model review were identified, and a more comprehensive set of calculations remains a topic for future studies.

The results of the modelling work utilizing MEWA and PORFLO were collected in a research report and a draft paper was written to be published in a peer-reviewed conference.

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 2011**

Currently MELCOR code is not able to consider aqueous species and how they affect the pH in the pools. In VTT ChemSheet has been coupled with MELCOR to calculate the equilibrium composition and the pH of the pools. The mass decay of significant radioisotopes are calculated using the ORIGEN2 code. The current ChemSheet model has to be tailored by hand for each MELCOR case and chemistry model separately which is time consuming process and susceptible to making errors. A new pool chemistry simulation model, ChemPool, is developed. It is a standalone program using ChemApp thermodynamic library directly for equilibrium calculation (which ChemSheet also uses internally).

Earlier advanced containment experiments (ACE) project shows that there might be the fission product release during corium and concrete interactions. The behaviour of fission products (etc. Sr, Ba, La and Ru) during corium/concrete interactions are investigated using GEMINI2 code modelling the ACE experimental data.

CHEMSHEET has been used in different several accident cases earlier (modelling fission products I and Cs chemistry and sump pH). To upgrade CHEMSHEET code it is important to have the same NUCLEA database that is used in GEMINI2. It is planned to utilize this database in CHEMSHEET models and develop a new CSFoam tool for viscosity calculations. The model is used to calculate the viscosity in all HECLA tests done in previous COMESTA project.

**Deliverables in 2011**

The modelled MELCOR accident scenario was a basic LOCA scenario and a break in the BWR main steam line. 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 and the amount of HCl released were calculated and pHs of water pools were calculated using a new developed ChemPool program. The formation of nitric acid in pedestal pool was 5.6 kg and in wetwell pool 8.7 kg. In water pool of pedestal most of the acid is coming from cables in form of HCl, 117.5 kg. Due this the pH drops rapidly to 2. The influence of nitric acid is small about 5 %. Also in wetwell the formation of nitric acid is small and the pH of wetwell pool stays alkaline due to CsOH. In these results only the formation of nitric acid in water phases were calculated.
The object of MCCI study was to model the ACE- experiments L1 and L6 by using the GEMINI2 software and NUCLEA database (NUCLEA-10_1.GEM). Thermodynamic databases have evolved much since the ACE-experiments were conducted. Under more specific examination are lanthanum, barium, strontium and ruthenium, which the models used in the ACE- experiments had particular problems with. The calculations made in the ACE-project (VANESA and CORCON models) predicted higher amounts of Sr, Ba, La and Ru than were actually measured. Gemini- calculations for the both ACE- test cases show that the amounts of gaseous Sr, Ba, La and Ru are very small or non-existent; the amounts of species actually observed in aerosol samples were slightly higher.

A thermodynamic system of selected elements and phases was created from NUCLEA database for calculations with ChemSheet tool in Excel. First melting temperatures of different concretes were calculated (from HECLA project experiments). Then melting behaviour of corium with different amount of steel and concrete were calculated at different temperatures. In the reactor pressure vessel the molten corium forms two immiscible liquid phases, the lighter oxide rich phase is formed on top and the heavier metal-rich phase on bottom. Then these hot molten phases will penetrate the steel wall of the reactor vessel and flow to concrete cavity causing concrete ablation. CSFoam tool was used to calculate the viscosity of this mixture at various temperatures and compositions. CSFoam tool contains correlations to calculate the viscosity, the surface tension and the density of molten oxide system. The calculation of the viscosity is based on the bonding state of oxygen in molten silicate and the flow mechanism of melts with a network structure. The needed viscosity parameters for uranium oxide (UO$_2$) and zirconium oxide (ZrO$_2$) were assessed and added to CSFoam tool.
2.5.3 Thermal hydraulics of severe accidents (TERMOSAN)

The objective of the work is to improve modeling capabilities on severe accident thermal hydraulics. MELCOR is Finland’s main severe accident analysis tool, and its thermal-hydraulic models are tested and validated with respect to passive containment cooling condensers. Two international OECD NEA research programs, THAI-2 and SERENA-2, are participated in the frame of this project. A master’s thesis on melt pool behavior in the reactor lower plenum is written.

Specific goals in 2011

A master’s thesis on core-melt behavior inside the reactor pressure vessel during the late-phase of a severe accident was written. The study brings the information obtained in the international experimental research program MASCA to the reactor scale. The tool used for the analyses is the integral code ASTEC (Accident Source Term Evaluation Code). The chemical equilibrium studies for core-melt were performed with ChemSheet. It was found that the initial chemical composition of the melt, depending on the accident scenario, has only a minor effect. However, the actual melt layer arrangement (Figure 2.5.3.1) was discovered to be the key point when considering the vessel failure and especially the place of rupture.

![Figure 2.5.3.1. Three possible melt pool configurations in the reactor lower head.](image)

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.

Modeling passive containment cooling systems with the MELCOR code was investigated. First, a literature review of passive containment condenser experiments was performed. Two experiments were selected for MELCOR 2.1 calculations. Simple experiments on condensation and aerosol deposition in a vertical tube were used for validation calculations of MELCOR’s condensation and deposition models. The calculated condensation rates were within 8% of the measurements. The calculated aerosol deposition was within 10 percentage units from the measurements.

PANDA T1.1 experiment, which investigated the ESBWR containment behavior in a severe accident, was calculated with MELCOR. It was found out that careful modeling of drainage of the condensed water in the passive containment condenser is important. With default treatment, the water formed tiny levitating pools inside the condenser. This caused numerical difficulties and serious oscillation to the results. Guiding the condensed water directly out of the condenser eliminated this problem. There was some uncertainty in the amount of helium injected to the PANDA facility. The calculations were made with 75 kg of helium. This amount was inferred from mass spectrometer measurements. According to the test report, 92 kg of helium was injected, but this appears to be an overestimation. It is concluded that MELCOR was able to calculate the condenser performance in the PANDA experiment with
7 % accuracy. This is sufficient for using the model in full plant calculations. The calculated pressure is compared with the measurement in Figure 2.5.3.2.

![Figure 2.5.3.2. Drywell pressure in PANDA T1.1 experiment. A comparison between measurement and MELCOR calculation.](image)

Finland participates in the OECD THAI-2 project, in which experiments are conducted on aerosol and iodine issues, hydrogen behavior, and HTGR graphite dust transport issues. In the frame of the TERMOSAN project, the first program review group and management board meetings were attended.

SERENA-2 (Steam Explosion Resolution for Nuclear Applications) is an international research program for investigating steam explosions, and it is coordinated by OECD NEA. It involves steam explosion experiments in the KROTOS and TROI facilities with real reactor materials. Within the TERMOSAN project, SERENA-2 meetings were attended.

**Deliverables in 2011**

- A master’s thesis on core-melt behavior inside the reactor pressure vessel during the late-phase of a severe accident was written. It was found that the melt layer arrangement is the key point when considering the vessel failure and especially the place of rupture.

- 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 iodine experiments in the THAI facility was given. A travel report was written.

- Modeling passive containment cooling systems with the MELCOR code was investigated. PANDA T1.1 experiment was calculated with MELCOR. MELCOR was able to calculate the condenser performance in the PANDA experiment with 7 % accuracy.
OECD THAI-2 program meeting was attended and a travel report was written.

OECD SERENA-2 program meeting was attended.

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 is 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 oxide 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 Scherrerr 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), OECD/NEA STEM and ARTIST2 programs.

Specific goals in 2011

The main goals in 2011 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 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 can be done more accurately.

Another goal was to study the behaviour of iodine, deposited as iodine oxide aerosol, on various containment surfaces. The objective was to determine the influence of radiation dose, humidity and temperature on the desorption of iodine from iodine oxide aerosol deposition on paint, stainless steel, copper, aluminium, zinc, palladium and platinum 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 third goal, an experimental facility (DIANA) was designed and constructed in order to experimentally investigate the particle depletion in a differentially heated cavity with turbulent natural convective flows.
The fraction of iodine desorbed [%] vs. Gamma radiation dose [kGy]

Figure 2.5.4.1. The fraction of iodine desorbed from iodine oxide particles deposited on painted surface at 20°C. The release of iodine was significant when samples were exposed to gamma radiation compared with the results when radiation was not used. The duration of the experiments was ~24 hours.

**Deliverables in 2011**

- In the primary circuit studies the source of iodine was CsI powder which was evaporated at 400°C, 550°C and 650°C on ceramic surface under Ar/H₂/H₂O 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, B₂O₃ or MoO₃, 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 is mostly in gaseous form, if some of the materials mentioned above were present.

- The iodine desorption experiments were conducted at 20°C, 50°C, 100°C and 150°C. The atmosphere was dry air. In some experiments the relative humidity of air flow was 80%. The paint was the same as used in Finnish NPP containments. Some of the samples were exposed to gamma radiation (Co-60). As a result, the enhancing effect of gamma radiation on the desorption of iodine from iodine oxide particles deposited on painted surface was measured at 20°C (Figure 2.5.4.1). Humidity in air flow seemed to increase the release of iodine as well.
The analysis of iodine oxide particles deposited on various containment surfaces was conducted with e.g. SEM + EDX, XPS, RAMAN, IR, and light microscope methods. Iodine was mostly in oxidised form on painted and stainless steel surfaces. Particles were identified to be I$_2$O$_5$. Whereas, iodine was mostly in non-oxidised form on copper and zinc surfaces. It seemed that iodine had reacted with the surfaces and formed probably copper iodide and zinc iodide.

The DIANA facility is a cube with side length 700 mm. It has a heated and cool side walls opposite each other and four glass walls. The glass walls are designed to be nearly adiabatic. Walls are heated / cooled using different temperature water. Particle Image Velocimetry (PIV) is used for measuring the fluid flow velocities (Figure 2.5.4.2). The laser is placed at the top of the cavity and the CCD camera in front of the front glass wall. PIV uses cross-correlation method to calculate the flow velocities inside the cavity.

Information related to the progress of the Phebus FP, ISTP, ARTIST2 and OECD/NEA STEM programmes was distributed.

### 2.6 Structural safety of reactor circuits

In 2011 the research area "Structural Safety of Reactor Circuits" consisted of six 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) and Water chemistry and plant operating reliability (WAPA).

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

The objective of the 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 and ageing of nuclear reactor materials. To meet these goals, crack initiation and its precursors are investigated using super slow strain tests in simulated NPP environments followed by detailed characterisation of the test specimens. Mock-ups and nuclear components are characterised to increase the knowledge of the characteristics of such material conditions. Understanding irradiation assisted cracking (IASCC) is increased through characterisation of irradiated stainless steel materials. The effect of environment on fracture behaviour is measured for nickel-based materials in hydrogenated water. The role of thermal ageing of different nuclear materials is explored. Fuel clad research capabilities are improved by development of testing capabilities for creep testing. Knowledge transfer to the young generation is performed through everyday mentoring, teaching, updating the digital report archive, and by giving the project personnel possibilities to learn by doing. The knowledge achieved in the project is utilised also in customer assignments. The latest international knowledge is brought to Finland through active participation in international projects and networks, and by delivering detailed travel reports.

Specific goals in 2011

The specific goals for 2011 were:

1. The role of deformation in the initiation of environmentally assisted cracking (EAC) in austenitic materials is investigated using super slow strain rate testing (SSSRT) of austenitic nuclear material in LWR environments. In 2011 the work was comprised of setting up the new testing facility for these tests, performing two demonstration tests, and characterisation of test specimens from earlier SSSR-tests.

2. Characterisation of nuclear components and mock-ups comprised in 2011 of characterisation of a low alloy steel – Alloy 52/152 – 316L stainless steel dissimilar metal weld mock-up received from professor C. Jung at KAIST, Korea.

3. Characterisation of irradiated stainless steels, where the work in 2011 consisted of characterisation of Type 304 stainless steel material irradiated to 7.7 dpa and after being subjected to a post-irradiation annealing treatments. A 2.9 dpa Ti-stabilised stainless steel core barrel bolt removed from a VVER after observed cracking dose was also characterised.

4. Investigations on the role of dynamic strain ageing on fatigue. These investigations were started in 2011 in parallel to writing a state of the art review on corrosion fatigue from a mechanistic viewpoint.

5. Influence of the strain rate and environment on fracture toughness properties of austenitic materials, where the work in 2011 consisted of performing fracture resistance tests in hydrogenated low temperature water and Alloy 152 material and comparative tests in pure water and air.
The anomalous rapid fracture of austenitic nuclear materials, observed in several laboratories, was evaluated based on available data.

Evaluation of the role of thermal ageing was started with preparation of the state of the art report.

Development of new test capabilities for creep testing was started in 2011 with preparation of the state of the art report on irradiation creep of fuel clad materials.

Knowledge transfer comprised in 2011 of preparation of a course on nuclear materials to be given at the Aalto university in spring 2012, by daily mentoring of new experts, by updating the digital report archive and by facilitating opportunities for learning by doing. International co-operation is an important part of the ENVIS project, both for bringing the latest knowledge to Finland, and to educate experts. In 2011 it comprises of active participation in selected meetings and conferences and delivery of detailed travel reports.

**Deliverables in 2011**

1. *Super slow strain rate testing of austenitic nuclear materials in LWR environments*

Due to overlapping need for the autoclaves used for SSSRT, a new test equipment for SSSRT was designed in 2010. Optimisation of control parameters with a new control unit was continued in 2011, performing a demonstration test with super slow strain rate. The work in year 2011 was affected by cleaning of the research hall, which interrupted all testing activities for the summer period.

Earlier SSSRT specimens made of Alloy 182 were characterised using FIB (focused ion beam), FEG SEM (field emission gun scanning electron microscopy), EBSD (electron back-scattered diffraction) and nano-indentation. The specimens, tested in simulated PWR environment and strained to about 5% plastic strain, revealed several short grain/dendrite boundary cracks. Some were most probably weld defects, others were located at regions with numerous precipitates, Figure 2.6.1.1, while some had no obvious discontinuity explaining the location of the cracks. Whether these very short cracks have a capability to grow with SCC mechanism remained as yet unresolved.

2. *Characterisation of precursors and mock-ups*

The characterisation performed on a LAS – Alloy 52/152 –SS 316L dissimilar metal weld mock-up revealed strong dilution of chromium and nickel, and enrichment of iron in the Alloy 52 butter, and to a lesser extent, in the Alloy 52 weld root. No major dilution was measured in the V-groove weld, where the composition was in accordance with the nominal composition of Alloy 152. A martensite-like layer was observed locally in the butter close to the LAS fusion line, Figure 2.6.1.2. High hardness values, up to 270 HV0.1, were measured in this area. The weld is also characterised in the SAFIR2014 FRAS project, in which the local mechanical properties are determined.
3. Characterisation of irradiated stainless steels

VTT has been characterising irradiated materials for the OECD Halden project for several years, increasing the knowledge and available literature data on radiation induced segregation and defect structures of different stainless steels. In 2011, the effect of post irradiation annealing on a 7.7 dpa 304 SS material was examined, complementing the results from 2010 that were carried out for a 24 dpa 304 SS. The findings in 2010 showed that a 400°C/6hr anneal had very little effect on microstructure, but at 500°C small defects were removed (Figure 2.6.1.3), some faulted loops grew, and fine precipitates (gamma prime) were mostly dissolved. The RIS profile was reduced by the heat treatments, but not totally removed. The 7.7 dpa material was examined in 2011 but these results will be evaluated and reported in 2012.
Figure 2.6.1.3. Post irradiation annealing at 400°C for 6 hours (left) did not reduce the amount of defects, while 500°C for 3.5 hours (right) resulted in a dramatic reduction in the radiation induced precipitate population.

A solution annealed Ti-stabilised stainless steel core basket bolt was removed, after observed cracking, from a VVER 440 plant. The dose of the material was only 2.9 dpa. A failure analysis has been performed earlier on the bolt, which revealed the cracking to be a consequence of intergranular stress corrosion cracking (IASCC and IGSCC). The FEG-STEM investigations performed in this project revealed clear grain boundary segregation with chromium depletion down to 13.5%. The silicon level on some grain boundaries was as high as 5.4%. High levels of phosphorous were measured at some TiC particles. The defect structure and density of the defects were in accordance with expectations for a stainless steel of that dose level.

4. Deformation behaviour under non-monotonic loading

Tests to investigate the possible role of dynamic strain ageing on fatigue were started using 304L stainless steel. Tensile tests were performed in air at 300 and 400°C with different strain rates and with and without prior cyclic loading. Differing mechanical behaviour during the monotonic tensile testing was observed at the two temperatures, when testing was performed in the DSA regime. More tests, as well as detailed investigations of the resulting deformation structures, are needed in order to fully understand the results. Parallel to the testing, a state of the art report focusing on the mechanistic aspects of corrosion fatigue was written, and open issues were identified. Tailored tests to investigate localized plasticity theories and underlying reasons for secondary hardening were some of the issues identified.

5. Influence of the strain rate and environment on fracture toughness properties of austenitic materials

The effect of hydrogenated PWR primary water on the Low Temperature Crack Propagation (LTCP) susceptibility of nickel based weld metals Alloy 182, 82, 152 and 52 has been studied for some years, and continued in 2011 on Alloy 152 material removed from the DMW sample received from KAIST/Korea. The fracture resistance test results show that the Alloy 152 material is somewhat susceptible to LTCP in hydrogenated PWR water (30 cm³/kg H₂O) at 55 °C. The fracture resistance values measured in hydrogenated environment are about 31% lower than the values measured in air and 57% lower than the values measured in pure 55 °C water.
6. Anomalous rapid fracture of austenitic materials in LWR environments

Anomalous fracture behaviour has been reported by several laboratories while performing load controlled crack growth rate tests. The rapid fracture occurs typically in cases where the tests have been running for very long times, and where the crack length has been underestimated and thus the stress intensity has been higher than the target level. Based on the scarce available data, the reason for the rapid fracture may in most cases be plastic instability, which occurs at high stress intensity in load controlled tests due to creep. However, more investigations to verify the reason for the rapid fracture, and the possible role of the environment, are needed.

7. Ageing of nuclear materials

Several areas of open issues were identified in the state of the art review on thermal ageing of selected nuclear materials, and the report will be used for planning of further research. Some of the issues identified are the possible mild sensitisation from manufacturing and consequent risk for low temperature sensitisation of LN-grade stainless steels, the risk for thermal ageing of new, high strength steel materials, the possibility of thermal ageing of Co-based materials and the possibility of short range ordering of nickel-based materials during long term operation. The influence of environment on thermally aged materials is a mostly unexplored area.

8. Development of test capabilities

The state of the art report on irradiation creep performed in 2011 reviews the latest information on irradiation creep for fuel clad materials. The review is used in the development of new test capabilities for creep testing of tubular materials, which will also serve the renewal of infrastructure.

9. International co-operation and education

International co-operation in 2011 consisted of active participation in the main international conferences, networks and projects on nuclear materials. The latest knowledge was brought to Finland and simultaneously many of the meetings were good educational experiences for young scientists. Knowledge transfer is also very important, and was executed through mentoring and learning by doing, not only in this project but also in customer assignments.

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 2011**

Specific goals for the project in 2011 were connected with four subprojects. Engineering assessment methods were developed with specific aim at implementation of promising extended finite element method (XFEM) in commercial FE-code Abaqus. Research on the applicability of the promising extended finite element method (XFEM) implementation in commercial FE-code Abaqus was started in the final year of the previous SAFIR2010 program by calibrating the underlying cohesive material model parameters for NPP materials. In 2011 verification and calibration of the cohesive model has been continued and real life examples have been studied. Available engineering tools and numerical software such as WARP3D suitable for fracture mechanics analyses will be studied and the applicability of these simplified and numerical tools in nuclear applications has been evaluated.

**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.

**Low constraint fracture** research aims at getting quantitative tools for evaluating criticality and growth of low constraint cracks. The results will be applicable to surface cracks and real irregular crack shapes in nuclear structures. Experimental work is a continuation for the low constraint international Round Robin performed in SAFIR2010. Analysis of the Round Robin will be finalised and a further test program planned and partly realised. Fracture resistance and toughness tests will performed using different low constraint cracked test specimens. Fracture behaviour of low constraint cases will be compared with deep cracked specimen behaviour. The constraint and biaxial effect and also their interaction in combination with the effect of warm pre-stressing (WPS) have been investigated.

Structural integrity analysis of DMW and procedures for fracture mechanical testing which could indentify the most critical zone of DMW for fracture will be reviewed and developed. Review of dissimilar metal weld structural integrity methods and their limitations will be made. Fracture mechanical and mechanical testing methods used in characterisation of DMW will be critically reviewed and suggestion about proper testing procedure will be presented. Existing DMWs available for testing will be collated. If these are not available, review and generation of relevant welding procedures for DMWs will be made in order to prepare DMW(s) for testing.

**Deliverables in 2011**

- The theoretical basis of XFEM was briefly presented and outlook on numerical software that implement XFEM was presented. Of the general purpose FE-codes, only Abaqus and Code_Aster provide direct support for XFEM routines. The capabilities of the XFEM
implementation in Abaqus were presented. Several elementary crack cases were analysed to determine the accuracy of the method and then some specialized cases often found in nuclear applications were calculated. It was found that static cracks can be evaluated accurately with reasonable effort but mesh refinement needs to be relatively fine in the crack region to obtain stress intensity factor over the full crack front. Special cases indicated that many complex scenarios can be studied with XFEM in Abaqus but the implementation still has restrictions that limit the possibility of utilizing the method in real life applications. The implementation is not yet as robust as other features in the software and simulated crack growth curves deviated from the reference results due to the limitations of crack tip singularity modelling.

When modelling crack growth, some kind of crack initiation and growth criterion has to be implemented. They often utilize stress or strain criteria for crack initiation and energy based criteria to govern crack growth. This kind of data is not readily available for many materials and often their correspondence to physical material properties is vague. Reliable utilization of the crack growth models thus requires extensive testing and calibration of the crack growth models.

Figure 2.6.2.1. Crack propagation as a function of applied loading. Displacements magnified by factor 5. Only half of the pipe is shown.
The LBB arguments are used in many European and other countries as a design basis of new plants and when evaluating the safety and integrity of existing piping components. LBB is particularly important for pipeline systems which are difficult to inspect. The most applied methods for evaluation of LBB are developed in USA, UK, Germany and France. The name of the method in Germany is break preclusion. Generally all the basic evaluation procedures are rather similar.

When performing a deterministic LBB analysis in Finland one can choose from a number of computational procedures and within reasonable limits provide values for the associated safety factors, as long as sufficient technical justifications are provided as well. Two case studies are examined.

The first case study deals with a circumferential crack in BWR piping. This case was studied in 1990’s, and here it was computed utilising the current methods. On the basis of the results, it can be concluded that the more refined present computational methods give more accurate results, which may be important, if the conditions regarding to LBB are justified.

![Computed leak rate for different crack lengths (2θ). The earlier results are shown as well.](image)

The second computed case is associated with the evaluation of elastic crack opening areas and stress intensity factor values for through-wall cracks located at the intersection region of a pipe branch connection. The results were produced for the benchmark study.
The constraint and biaxial effect and also their interaction in combination with the effect of warm pre-stressing (WPS) have been investigated in the European CABINET project which has been realised as a part of FAR. In 2011, newer, well documented test results were compiled and further analysed for further verification of the proposed engineering tools for integrity assessments, the focus being on how the aforementioned phenomena and loading conditions should be handled in these assessments. Besides the recent data, a few well documented older data sets were collected are described.

Figure 2.6.2.4. Beneficial warm prestressing has not been eroded with irradiation.
• Specimen cutting and prefatigue for a relevant DMW was performed for fracture resistance determinations.

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

The objectives 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 and to verify the reliability of NDE simulations.

Non-destructive testing 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 shutdown period. It would be very useful to find also possible failure precursors i.e. microscopic material changes at that point. It is also 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.

Specific goals in 2011

Specific goals in 2011 included the understanding of different indications in ultrasonic inspections. Different artificial reflectors were made into the reference specimens. Some physical properties of the reflectors can be considerably different with properties of the real defects. The ultrasonic behaviour of the reflectors and their ability to simulate real defects was one of the goals.

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) curve provides valuable data for the reliability of the used non-destructive testing method. The calculation of a POD -curve requires dozens of measurements from defects that represent well the parameters (size, shape, angle, roughness, etc.) of service-induced defects.

Microscopic imperfections in structural materials cannot be observed by traditional ultrasonic NDE since they are far smaller than the ultrasonic wavelength. An introduction report to the state of the art of nonlinear ultrasonics in NDE was written. Another state of the art study is done for the digital radiography. Only during the last decades new digital detectors have appeared. The current state of the digital radiography in the field of the nuclear industry is reported.

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. Eddy current methods have been under development for measuring the thickness of the magnetic pile.
Figure 2.6.3.1. Indications of the EDM notches and thermal fatigue cracks are clearly visible.

Figure 2.6.3.2. The script chart presentation of ET signals due to magnetite piles of different thickness under the test tube. Up to 15 mm the vertical amplitude of the ET signal grows with the increasing thickness of magnetite pile. The applied ET frequencies were 10 kHz and 25 kHz.
Deliverables in 2011

- In 2011 several fatigued samples were inspected using C-mode scanning acoustic microscope (SAM) and one sample with thermal fatigue cracks and similar size of EDM notches was inspected using conventional ultrasonic equipment and SAM. The results showed that the amplitude of the indication of the EDM notch is bigger than the same size fatigue crack (Fig. 2.6.3.1.).

- Simulated ultrasound responses of the cracks are calculated with CIVA software. Probability of detection curve is calculated using the calculated amplitude data and the defined detection limit. Example calculations for cracks, resembling mechanical fatigue cracking, have been done.

- A thorough literature survey of the nonlinear ultrasonics has been accomplished, containing the study e.g high amplitude ultrasonic pulses. Recent studies show that in certain circumstances low amplitude ultrasonic pulses generate a much stronger nonlinear response than expected. There are evidences that water in tight cracks may not behave like bulk water. Confined water exhibit more solid like mechanical behaviour.

- The thickness of magnetite piles on the tubes that were simulating the horizontal steam generator tubes have been sized using eddy current techniques and ordinary bobbin probes (Fig. 2.6.3.2).

- A report that reviews recent literature of the digital radiography in the field of the power generation industry has been written. New research scientist has familiarized himself with radiography along this work.

- Education of the research scientists continued, including also courses of specific non-destructive evaluation methods. Five SFS-EN 473 NDT level 2 certificates have been reached in 2011. (Tarja Jäppinen ET, Antti Tuhti RT; UT, MT and PT)

- Five research institute reports have been completed.

- Three World NDT conference papers accepted

2.6.4 RI-ISI analyses and inspection reliability of piping systems (RAIPSY)

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 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 2011

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. Further, the aim was to publish the recent licentiate thesis by one of the project group members in the VTT Publications series.

Another task of the first subproject was to examine the applicability of the NPP piping degradation databases for producing/improving probabilistic crack estimates needed in the probabilistic fracture mechanics (PFM) based simulations with VTTBESIT. Of the current notable databases, the main source was the OECD Piping Data Exchange Database (OPDE).

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 on Risk. ENIQ TGR develops recommended practices and discussion documents related to RI-ISI. Further, the aim was send a manuscript on the results of the NKS project from the last year of the corresponding SAFIR2010 RI-ISI related project to a peer reviewed scientific journal.

Another task of the second subproject was to examine the effect of initial flaw and load assumptions on risk estimate changes. The duration of this task is planned to be two years, and during the first year the aim was to collect all necessary technical background data and information, perform some computational analyses as well as to provide some preliminary/tentative conclusions. Further, the aim was to have published in a peer reviewed scientific journal an article concerning the assessment of size distributions of initial cracks nucleating during BWR operation due to stress corrosion cracking (SCC).

The third subproject covers other international co-operation besides participation to ENIQ TGR activities as well as project management. In addition to the latter issue, the aim was mainly to participate in RI-ISI related workshops and NULIFE Network of Excellence activities.

Deliverables in 2011

- The results of the research work on structural reliability analysis methods focus on further development and application of a more realistic approach for crack growth computation with analysis tool VTTBESIT, and on a study of sampling methods used in probabilistic crack growth computations. The purpose of these efforts was to improve the probabilistic degradation potential computation capabilities needed in risk informed in-service inspection (RI-ISI) analyses.
The study on the applicability of the NPP piping degradation databases for estimation of crack initiation and leak frequencies in NPP primary circuit components resulted in collection of relevant data, implementation of computational procedure and two case studies. The targets of application in these studies were primary circuits in both BWR and PWR plants. The applicability of Database OPDE to the analysis purposes in question was reviewed and tested.

The recent licentiate thesis “Structural lifetime, reliability and risk analysis approaches for power plant components and systems” by one of the project group members was published in the VTT Publications series as P775 in December 2011.

The VTT contributions to ENIQ TGR documents included those on new plant RI-ISI and reasonable risk reduction achieved through RI-ISI.

The results from the first part of the study on the effect of initial flaw and load assumptions as well as of inspections on risk estimate changes mainly concern failure potential analyses with probabilistic VTTBESIT code and two Markov process applications for three representative NPP piping welds, covering a wide range of initial flaw and load assumptions. In this first phase, the effect of inspections was excluded from the analysis scope.

A scientific journal article “On assessment of initial cracks for RI-ISI analysis purposes” was published in August 2011 in Journal of Materials Science and Engineering (JMSE).
2.6.5 Advanced surveillance-techniques and embrittlement modelling (SURVIVE)

The SURVIVE project has three separate objectives, namely 1) validation of small specimen test techniques, 2) development of multiscale modelling and 3) identification of irradiation induced microstructure and correlation of microstructure with irradiated material properties. Main focus of the works is on the behaviour of VVER440 welds. Small specimens are utilized widely in radiation effect studies but validation of the methods has not been properly performed. In multiscale modelling the multicristalline nature of metals has been taken into account instead of assuming quasi-homogeneous materials. In microstructural characterisation work co-operation with other research institutes is widely utilized. Own work is focussed on resistivity measurements.

Specific goals in 2011

Specific focus in 2011 in the small specimen technique was on reconstitution technique. Specimen deformation and heat transient in the welded specimens were characterised to sufficient extent and the measuring capacity of a reconstituted specimen as a function of centre insert length was determined for some materials. Correlation of lateral expansion measured in a Charpy-V test was derived with the axial extension of deformation with large amount of tested archive specimens. Figure 2.6.5.1 shows the data for the validation of insert length with JRQ-material. Enough data required to estimate the acceptability of reconstituted Charpy-V upper shelf specimens has been created. The work has been reported.

![Figure 2.6.5.1](image.png)

*Figure 2.6.5.1. The blue points are the impact energy values a function of the insert length. The green points are the valid values derived from the instrumented graph as described in Figure 2.6.5.2. Incorrect impact energy values will be measured, when the insert length is ≤ 12mm.*
Figure 2.6.5.2. Integrated energy from an instrumented graph is given as a function of specimen deflection. The graph is made for 8mm inserts and it shows the points were the integrated reconstituted specimen graphs start to deviate from the full length specimen graphs.

The validation work on acceptance of small CT-and 3PB specimens was not started due to unavailability of test equipment and loading devices.

In multiscale modeling a library of polycrystalline structures was created and reported and it will be used in further modeling.

In microstructural work resistivity measurements were performed with 501 weld material in I-, IAI- and IAIAI-conditions. Single measurements are repeatable, i.e. same values are gained in repeated measurements. The measurements were performed at liquid nitrogen temperature and hence no noise due to temperature variation is expected. The specimens are heated in vacuum oven for relatively short periods between the measurements. It seems that the thin oxide layer formed on sample surface even, if removed by gentle polishing, will cause additional noise in the measurements. One reason for the noise may also be the variation of the contact point on sample surface from measurement to measurement. The current data do not allow detailed interpretation of annealing behavior. New specimen geometry has been developed including cold plated contact surfaces. Example of current data is shown in Figure 2.6.5.3.

Microstructural samples of VVER440 welds were prepared and shipped to HZRD (Helmholz Zentrum Rossendorf-Dressden) and to CIEMAT. HZRD will perform SANS measurements in spring 2012 and CIEMAT TEM work on these samples.

ATOM probe and PA tests on samples sent to Tohoku University couple of years ago are currently under measurement. Data is expected to be available in spring 2012.
Figure 2.6.5.3. The annealing curves for three different material conditions. The data clearly includes random scatter, which is aimed to be removed with new specimen geometry.

Deliverables in 2011

- Large enough data base has been created for estimating acceptability of reconstituted Charpy-V upper shelf tests. The data includes specimen deformation data, temperature transient data, characterization of weld seam properties and determination of anneal zone near the weld. The work has been reported.

- A subprogramme required in multiscale modeling has been developed, tested and reported.

- Resistivity data on annealing behaviour of varying irradiation conditions is available but further reduction of scatter is required in order to identify reliably the annealing mechanisms. Improved specimen geometry and contact points (gold plating) have been preliminarily verified.

2.6.6 Water chemistry and plant operating reliability (WAPA)

The main objective of the project is to increase understanding on the influence of water chemistry on plant operating reliability. Pre-oxidation of component and system surfaces can have a major effect on the corrosion rate and activity incorporation onto the surfaces. Knowledge on optimisation of pre-oxidation techniques is needed when new systems are taken into use for the first time and when new or decontaminated components and/or system parts are taken into use. The objective in this area is to develop tools to determine on-line the quality of pre-oxidation and to apply the tools in optimising the water chemistry for pre-passivation of new plants (e.g. Hot Conditioning). Deposition of corrosion products onto component surfaces is a key process e.g. in crud formation on fuel cladding and sludge formation in steam generators, both of which may have safety related consequences. This
The project aims at developing understanding on the effects of surface chemistry, water chemistry, hydrodynamical conditions and thermal gradient on deposition processes.

**Specific goals in 2011**

The primary circuit of a new 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 the future power cycles, and thus minimise the activity build-up at the plant. Preoxidation of system and component surfaces will be of importance e.g. when deciding on the HFT procedure for future nuclear power plants. In the future also larger system parts will be decontaminated and possibly peroxidised before taking into use.

![Figure 2.6.6.1. Effect of lithium (Li) and boron (B) on the corrosion resistance ($\Omega \cdot \text{cm}^2$) of Inconel 690 steam generator tube. Exposure to water at $T = 292^\circ\text{C}$ and with dissolved hydrogen, $[\text{H}_2] = 30 \text{ cc/kgH}_2\text{O}$.

There is no international consensus on the best available procedure for HFT. Open questions exist on the minimum length of the passivation time, the optimal concentration of lithium (Li as LiOH), and the use of boron (B as boric acid $\text{H}_3\text{BO}_3$). In this project an on-line electrochemical method (based on electrochemical impedance spectroscopy, EIS) to monitor the passivation process was and verified for carbon steel in co-operation with BARC, India. In 2011, the method was used to study the effect of Li and B on passivation of Inconel 690 steam generator tube. The results shown in Figure 2.6.6.1 indicate that the higher the Li-concentration the lower the corrosion resistance. Boron addition is beneficial at all Li-concentrations, although the effect is more pronounced at lower Li-concentrations.
Regarding the deposition of crud and sludge, the goal in 2011 was to study through literature the effects of pressurised water reactor (PWR) shut-down and start-up procedures as well as the effect of zinc (Zn) injection on the source term for deposition. On the experimental side the goal was to develop an experimental facility enabling measurement of the zeta-potential of magnetite as a function of PWR secondary side water chemistry.

### Deliverables in 2011

- A range of experiments were carried out in a flow-through measurement cell at \( T = 292^\circ C \) and \( p = 100 \) bar, studying the effect of lithium and boron on the passivation of steam generator tube material Inconel 690. A linear negative correlation was found between the corrosion resistance and the lithium concentration, i.e. the higher the Li-concentration the lower the corrosion resistance. Addition of boron was found to be beneficial especially at lower Li-concentrations. The results were communicated in one VTT Research report and one scientific journal publication.

- In PWR secondary circuit, deposition of magnetite into steam generator is causing severe corrosion and tube cracking problems. The facility developed for measurement of the effect of water chemistry on surface charge (zeta-potential) of magnetite as a function of temperature was utilised within a MSc thesis work. The experimental results from comparison of morpholine, ammonia and ethanolamine at room temperature show that magnetite surface charge can be considerably influenced by the choice of water chemistry.
A high surface charge tends to keep magnetite in colloidal form (non-flocculating) so that a larger part of it can be removed from steam generator through blow-down thus avoiding deposition. The results were communicated in the form of MSc thesis.

- The use of zinc injection in PWR’s for minimisation of corrosion is wide spread and shown to be effective. Experience from injecting zinc already during the Hot Conditioning stage of a new plant is very positive, resulting in a 40% reduction in the activity build-up during the following power cycles. The start-up and shut-down procedures of a PWR plant have a major effect on the activity build-up on primary side component surfaces. These results were communicated in the form of two literature studies published as VTT Research reports.

2.7 Construction Safety

In 2011 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 an 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. The target used will be force plate or concrete wall with reinforcement and liner. The wall can be designed with pre-stressing bars and liner and later on also curved structure or floor-wall structure. The test results will also be used for numerical analysis in SMASH project in SAFIR2014.

The impact project includes 9 partners from Europe and America which are funding the project or giving information of earlier impact tests. The specific objectives described above were specified in further detail. These details are clarified subsequently in the description of the tests.

Specific goals in 2011

1 Improving of test apparatus

The test apparatus has been modified to be suitable for liner and curved walls. The tests of floor-wall structures will be done later in 2012. This will be done by redesigning the construction of the supporting frame and back pipes. The front part of acceleration tube was slightly damaged after 100 shots and has been repaired or strengthened. The acceleration tube has been modified for more accurate shooting to middle point of the wall.

_The object of the task 1_ was to improve the test apparatus to be suitable to test liner, curved and floor-wall structures. The acceleration tube has been repaired.
2 Improvement of measuring system

The data acquisition system will be improved by increasing measuring devices, which are capable to measure forces and stresses when floor-wall structure is used as target. This will be done later in 2012 by designing a new force measuring system under the structure to be tested. The permanent deflections of the wall has been measured by laser and deflection sensors before and after the test. New measuring card including 8 channels has been obtained for new test types. The permanent strains of rebars will be measured by fibre optics, the system has been designed and tests have been begun.

The object of the task 2 was to improve the data acquisition system and number of sensors to get more accurate and reliable test results.

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 object of the task 3 was to ensure the tests will succeed in a desired way and also the number of unsuccessful test is minimized.

4 Testing of missiles, walls and structures

The main purpose of test campaign in 2011 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 also by simplified aeroplane model with wings and engines using force plate or concrete wall. In 2011 new concrete wall tests with liner at the back face has been performed. The test matrix has been decided by TAG meeting.

The object of the task 4 was to test stainless steel missiles and rigid missiles onto the concrete wall. Also some wall with liners has been tested at the end of 2011.

5 Archiving of test results

All the drawings, places of sensors, material test results, videos, photos and measured data e.g. has been saved systematically to VTT DOHA system to be distributed to all partners. Some tests (missile and concrete structure tests) have been post-analyzed 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 t analysis has been done by LabView program. The data will be used for FE analysis in SAFIR2014/SMASH project.

The object of the task 5 was to archive all the results systematically to DOHA system and also to analyze test results to be able to decide if the measuring devices have been worked and the sensors have located in a proper locations.
Figure. 2.7.1.1. Concrete wall, thickness of 150 mm (left) and stainless steel missile (right) after impact test.

Deliverables in 2011

Improving of test apparatus

- A new frame for concrete walls and pre-stressed concrete walls has been used in the tests since 2008. The back part of the apparatus has been renewed and strengthened.

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.

Pre-calculation of the tests

- All the tests including new liner tests have been pre-calculated to have successful tests.

Testing of missiles, walls and structures

- In 2010 - 2011 fifteen (15) concrete walls, thickness of 250 mm have been tested successfully using different impact velocities, pre-stressing levels and reinforcements. The main idea was to research the effect of pre-stressing level, liner effect and failure modes with different parameters.

- In 2010 - 2011 seventeen (17) concrete walls, thickness of 150 mm have been tested successfully using different impact velocities and reinforcements. The main idea was to research soft missiles with different velocities.

- In 2010 - 2011 twenty one (21) force plate tests have been tested successfully using different impact velocities. The main idea was to research force time function and spreading of the fuel with different velocities.
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.

Impact project, phase 2 with foreign partners will end at the beginning of 2012, but the new continuation project phase 3 has been decided to establish. The negotiations are going on.

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 aging management, which supports many analysis and design tools related to the management of concrete structures in nuclear power plants. Using the same harmonised data system many analyses related to the safety, performance, and service life management can be conducted. Several tools used in the analyses are developed during the project. By the aging 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 SAFIR2010/ SERVICEMAN project is linked to the platform and the new database system.

MANAGE is a joint project with VTT and Aalto University.

Specific goals in 2011

Specific goals for 2011 were the following:

- A working plan on the MANAGE platform,
- Plan for the Central Database,
- Plan for interfaces between the Central Database and the analysis tools,
- Plan for 3-D visualisation of structures,
- Plan for integrating the life cycle design program ServiceMan to the system with special focus on new NPP units,
- Plan for Inspection Database,
- Condition analysis based on samples taken from the cooling water channels of existing power plants,
- Corrosion measurements in Loviisa 1 cooling water channels,
- Participation in the activities of OECD/NEA IAGE concrete work group and other international cooperation,
- Improvement of the structural analysis of OL2 containment building.

**Deliverables in 2011**

A working plan for the aging management System for concrete structures in NPPs was produced (VTT + Aalto). The system consists of a platform with both internal and external tools for various purposes of aging management. An integral part of the platform is the Central database and the Graphical user interface. In the report the layout, interfaces and various tools of the system have been described on conceptual and technical levels. A scheme on the MANAGE platform is presented in Figure 2.7.2.1.

![Platform for the aging management](image)

**Figure 2.7.2.1. Platform for the aging management.**

The Central Database will be divided into six branches (Figure 2.7.2.2): Inspection database, Service life management, Visualisation, Monitoring, Structural analyses and User management. The Central database is a relational Oracle database.

![Central database](image)

**Figure 2.7.2.2. Central database.**
Users can have access to the database by using Oracle graphical user interface or through an implementation application. Sensors transfer data directly to the database through a special interface and the implementation application can produce files for the use of Structural analysis tool and the Visualization tool. Implementation application contains also internal interfaces between the Central database and the Service life management tool ServiceMan and other possible Excel tools. In the report a detailed description on the databases of ServiceMan has been given for possible new power plant units.

The interface for visualisation will enable users to download and open CAD files (e.g. DWG, DXF, DGN) using the links that are stored in the database. The universal visualization tool, AutoVue, which is able to read most of the existing CAD standards will be used. Similarly, as the visualization data, the data for structural FE solvers will be provided in its native format (e.g. Abaqus CAE file or input file INP).

The Inspection database is part of the ManAge central database which includes all the NPP condition survey data. The inspection data is translated to an electronic form and then stored in the ManAge inspection database. The physical inspection database model describes all table structures, including column name, column data type, column constraints, primary key, foreign key, and relationships between tables.

Condition assessment of cooling water channelling was performed based on concrete samples. The samples were taken from OL2 and LO1. The chloride contents in LO1 channels were much higher than in OL2 and exceeded in many places the critical chloride content. The corrosion rate measurement results were contradictory with visual observations. As a rule the measurements indicated corrosion although no sign of corrosion was visually observed.

International cooperation is necessary for a successful aging management system. The annual OECD/NEA IAGE concrete group meeting was held on 6th and 7th April in Paris. An OECD/NEA IAGE Expert Meeting was held on 20th and 21nd April, 2011, Villeurbanne, France, hosted by EDF. The topic was “Post-tensioning methodologies for containment buildings: greased or cement grouted tendons – consequences on monitoring, periodic testing and modelling activities.” Several people from Finland took part in the Expert Meeting. Olli-Pekka Kari (doctoral student in Aalto University) visited Hokkaido University 28th Feb. – 21st Aug. 2011.

Structural analyses on the OL2 containment were successfully conducted with the improved FE model. The improvements of the model were: a) a more realistic stress distribution along tendon length, b) consideration of long term losses in the tendon forces, and c) calculating with both linear and nonlinear material properties. A part of the task was done by ÅF-Consult 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.

An aircraft impact on safety related structures, in spite of its low probability, has for a long time been recognized as a relevant loading case, especially in designing plants to areas with
heavy air traffic. It is required (Government Decree 733/2008) that the nuclear plant design takes into account large airliner crashes, as well as fires and explosions. Structural analyses of these phenomena require nonlinear numerical analyses. In order for the results of these analyses to be reliable, the applicability of used methods should be verified by combined experimental results and analytical methods.

**Specific goals in 2011**

Soft and hard missile impacts: State of the art report was compiled on existing methods, information gained from IRIS2010 OECD benchmark work was utilized too. Relevant model parameters (boundary conditions, material, loading etc.) were identified and sensitivity studies for selected parameters were carried out. Simplified methods were developed further. Load function prediction was improved by applying an analytical folding model.

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.

![Figure 2.7.3.1. Loading functions by Riera method and folding mechanism](image)

**Deliverables in 2011**

- State of the art on available analyses method for impact loaded reinforced concrete structures.

- Analytical model for folding mechanism developed. The preliminary results are reported in a conference paper.

- Pressure loads due to varied TNT explosions were calculated. Structural calculation models were developed and tuned based on the available experimental data.

- FDS simulations on selected tests have been carried out. Spray front speed comparison have been performed for 6xx –series and drop size comparisons for FPx –series.
Compilation report on liquid results in selected wet missile IMPACT tests has been completed. The experimental findings of liquid study of all wet missile IMPACT tests so far have been summarised.

Figure 2.7.3.2. FE model of the slab and missile with detail of the model on the right.

Figure 2.7.3.3. Overview on the flame ball and smoke plume.
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 tasks 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 organizations working with issues related to seismic engineering.

Specific goals in 2011

The first objective in 2011 was to identify the most pressing needs to be focused within the Subprojects. We aimed to understand: (i) what is known internationally, (ii) what needs further consideration because it might be special in Finland compared to the international experience and (iii) what are the generally accepted problem spots of the seismic assessment and qualification process of NPP’s. The second goal for 2011 is to define the content, and prepare the teaching material for the educational tasks of the project.

The fulfilment of these goals requires the carrying out of the state-of-the-art review in each subproject, and the systemization of the findings. In terms of hazard the goal was to produce a state of the art summary of the understanding of seismic hazard in Finland. This work introduces the previous studies conducted in Finland, and some comparative studies from regions with similar seismicity.

![Figure 2.7.4.1. Priorities in the design process for ordinary buildings and NPP’s, as dictated by the followed performance objectives](image-url)

*Figure 2.7.4.1. Priorities in the design process for ordinary buildings and NPP’s, as dictated by the followed performance objectives (Increasing font size means more important priority)*
For structural assessment it was proposed to review the current design method for NPP and evaluate the detail level of the prescriptions in design codes (ASCE 4-98, EN 1998). In 2011 important effort was directed also to the preparation of the teaching material for the courses.

**Deliverables in 2011**

- The collection of information on seismic hazard and risk assessment, both literature and review of software has been carried out by the Institute of Semiology and ÅF-Consult. Also, the present-day projects and programmes worldwide about seismic hazard and risk assessment have been reviewed and their relevance to conditions in Finland assessed.

- The earthquake databank containing selected seismic events from Fennoscandia has been prepared, with the purpose of forming the basis of seismic hazard and risk assessment. Approximately 100 earthquakes, rock bursts and mining induced seismic events from Finland, Sweden, Norway, Baltic Sea, Bothnian Bay and Kola Peninsula area are collected. Magnitude range is 1.9-5.0 (MLHEL). The basis for earthquakes to be chosen to the bank is their size and areas (geographical/geological/tectonical environment) of hypocentres and epicentral distances near the stations. Most events have epicentral distances under 100 km from the nearest station. The events have occurred in depth of 0-30 kilometres.

- Concerning structural assessment methods we reported a state-of-the-art overview of accepted seismic design methods for the nuclear field versus those for ordinary structures. The differences in seismic design philosophy are noted between the two fields, and detailed comparison of the design standards ASCE 4-98 and EN1998 was carried out. Generally speaking, ordinary building design is targeting the objectives of life-safety or non-collapse, but in NPP’s the goals are to have installations fully operational or perform a safe shutdown procedure (operational after inspections). The different performance objectives result in different analytical tools and design priorities being employed in the two cases (Figure 2.7.4.1). A set of recommendations to be considered for seismic design of NPP facilities in Finland was made with the intention of improving the YVL2.6 guide.

- The course for the Earthquakes and seismic loading on Structures (Rak-43.3160) has been prepared in 2011 and is underway since 27.01.2012 in Aalto University. The success of the topic is reflected by the approximately 60 participants enrolled in the course. The 14 public lectures are a unique cooperation framework between Aalto University, The University of Helsinki and VTT, and have the strongest direct impact from among the deliverables of SESA in 2011.

**2.8 Probabilistic risk analysis (PRA)**

In 2011 the research area "Probabilistic Risk Analysis (PRA)" consisted of three projects: Extreme weather and nuclear power plants (EXWE), Risk assessment of large fire loads (LARGO) and PRA development and application (PRADA).
2.8.1 Extreme weather and nuclear power plants (EXWE)

The overall objective of the research is to produce a comprehensive study about 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 the Finnish nuclear power plants, as well as the effect of climate change on these. Some exceptional weather events or combination of different events may prevent normal power operation and simultaneously endanger safe shutdown of the plant. Extreme weather events could affect, for example, the external power grid connection, emergency diesel generators, ventilation and cooling of electric and electronics equipment rooms and the seawater intake.

The sea level scenarios can be used as guidance when the design basis of the new nuclear power plants are defined. Simultaneous flooding of several safety important compartments must be practically eliminated. As the design criteria are defined in the near future, updating the sea level scenarios taking into account the latest scientific knowledge is a first-priority issue. The scenarios are also applicable in other areas of the society, like planning of coastal constructions and preparedness for natural hazards. The results on extreme weather phenomena can be used in defining the safety regulations for nuclear power plant constructions and operations.

Specific goals in 2011

The sea level is rising due to thermal expansion of sea water and melting of land-based glaciers. The highest sea level rise scenarios contain a large contribution from the Greenland and West Antarctic ice sheets. The West Antarctic ice sheet is more critical from a Finnish point of view than Greenland – which was studied in SAFIR2010 - since the melting of large ice masses results in changes in the gravitational field of the Earth, resulting in meltwater from the Antarctic ending up to the northern hemisphere. The effect of potential melting of the West Antarctic ice sheet on sea level was thus studied in 2011.

Exceptional heat waves may influence nuclear power plant safety systems in several ways. The ventilation and cooling of electronics rooms may be deteriorated. During a long heat wave the rise of the seawater temperature may deteriorate the function of heat removal chains to sea. In addition, algae growth and biological fouling may increase and lead to clogging of seawater systems. In the worst case these phenomena result in the loss of some safety functions.

Accompanied with human-induced increases in mean temperature, the frequency and intensity of heat waves are expected to change as well. The occurrence of high temperatures in Finland was previously examined within SAFIR2010 based on observations until the year 2008, however, not taking into account potential non-stationarity of the observational time series due to the on-going global warming. The occurrence of heat waves was thus studied in 2011, based on observed historical time series as well as climate model simulations for the future.

Deliverables in 2011

- The potential melting of the West Antarctic ice sheet and its impact on the Finnish coast was studied based on a review of published literature. The review includes an overview of the properties of the ice sheet and mechanisms that could disintegrate it. The observed changes in ice flow from the glaciers were reviewed, as well as future scenarios. The
The effects of thermal expansion, melting of glaciers, local land uplift and wind-induced changes in the Baltic Sea level were combined to update the scenarios for the sea level on the Finnish coast (Figure 2.8.1.1). The 124-year long observed sea level time series, as well as climate model runs of the IPCC Fourth Assessment Report were utilized. The resulting scenarios, their scientific basis, as well as an assessment of different factors affecting them and their uncertainties, were collected in a scientific article (Johansson et al., 2012) and submitted to an international peer-reviewed journal.

Observational time series of air temperature were analysed in order to detect potential past trends and other modes of non-stationarity in the occurrence of summer and winter heat waves. The observational database used in the project also covered the record-breaking summer 2010. The small portion of statistically-significant positive trends in observed summertime high temperature events suggests that the warming experienced so far in summer has been relatively small compared to the large natural inter-annual and inter-decadal variations of climate in Finland. This finding supports the results obtained previously by Saku et al. (2011) for high temperatures, as these were calculated assuming no trends. For winter temperatures, trends in time need to be taken into account even when carrying out extreme value analysis of the past climate, as demonstrated in a report by Jylhä et al. (2012).

Climate scenario data on hourly basis, created in a parallel research project REFI (Test reference years for building physics and impacts of climate change) was utilized to develop scenarios of recurrence of heat waves in Finland by around the 2050s (Figure 2.8.1.2). The results were presented in the report mentioned above and summarized in a conference paper by Tietäväinen et al. (2012). The conference paper summarized the different external geophysical risks (extreme weather and sea level phenomena), probabilities of their occurrence and impacts on nuclear power plant safety.

Figure 2.8.1.1. Sea level scenarios for Hanko and Vaasa; average scenario (solid line) and low and high estimates corresponding to the 5% and 95% cumulative probabilities (dashed lines). The observed annual mean sea levels in 1890–2010 are also shown.
Figure 2.8.1.2. Projected change in time of the annual maximum duration of spells with air temperature higher than 23°C (left), 27°C (middle) or 31°C at four sites in Finland (from left to right: Vantaa, Jokioinen, Jyväskylä, Sodankylä). The results for 1980-2009 are based on observations, while those for future 30-year periods centred on the year 2050 are based on model projections, assuming the SRES A2 emissions scenario. In each column of values are shown (from bottom to top) the minimum, the 10th, 25th, 50th, 75th and 90th percentiles and the maximum during 30 years.

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 fulfilment 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 2011

In 2011, new computational techniques were developed to evaluate the physical separation in fires resulting from the ignition of large fire loads. The elements of the defence-in-depth principle in fire protection were specified in the context of the risks associated with large fire loads at Finnish NPPs. The methods to evaluate the fulfilment of defence-in-depth principle were studied, including the development of new computational capabilities for the interoperability of fire simulations and structural analyses. This includes a new interoperability tool for the coupling of CFD fire simulations and FEM simulations of structural elements.

Concerning the response of the plant functions and personnel to fire, a survey of the effects of heat and smoke on digital automation was made. Also, the existing Fire-HRA method (Figure 2.8.2.1) was developed and applied on TVO cable tunnel/room scenario for realistic demonstration and testing.

Another goal was a validated capability to simulate large fire loads and to maintain the open-source simulation software, developed under international co-operation. The new capabilities will the prediction of liquid pool fire burning rates and the description of cables as sub-grid-scale elements of CFD fire simulation. Additionally, the Finnish participation in OECD PRISME and PRISME2 projects was planned through LARGO project.

Figure 2.8.2.1. Stochastic operation time model for personnel’s response to fire.

Deliverables in 2011

- A proposal was made for the method to be used for the assessment of fire defense-in-depth capability of NPP spaces. A modified and improved version of the French EPRESSI
method will be developed and applied during the following years. The method can be used to utilize the test based fire resistance classifications in room specific assessment.

- A literature review was carried out on the effects of smoke and heat on the components of digital automation systems. Very little new information was found, and it seems that the experiments that were made at Sandia National Laboratories in the end of 1990’s and early 2000’s, still represent the best knowledge. It was also found out that the U.S.NRC has started new test campaign on the same topic at the National Institute of Standards and Technology (NIST). Discussions with the NIST researchers were initiated concerning the possible co-operation in this field.

- The TVO cable room Monte Carlo fire simulations were finished successfully. (This study may hit the world record in the amount of computer resources consumed for a single fire problem.) The results were analyzed and combined with the stochastic operation time model to calculate the probability of second sub-system failures in case of cable-originated fires. The results will be published in PSAM-11 and also in a journal article.

- The performance of two liquid surface evaporation models in the CFD fire model was investigated using the OECD PRISME experimental data as a basis for the validation. The starting level of the PRISME simulations was determined by calculating the validation metrics (Figure 2.8.2.2) in case of specified fire heat release rates.

- The methodology for the description of burning cables as sub-grid-scale elements of CFD fire model was developed. Experimental characterization of chemical, heat-releasing reactions in a multi-component material, such as a PVC cable material, was developed using test data from U.S.NRC test campaign CHRISTIFIRE.

![Figure 2.8.2.2. An example of validation summary data for the FDS simulations of OECD PRISME experiments.](image)
2.8.3 PRA development and application (PRADA)

The objective of the project is to develop methodologies that are applicable in the risk analysis of nuclear power plants, and to increase the applicability of nuclear power plant risk analyses. Due to the cross-disciplinary nature of PRA, the PRADA project also consists of subtasks with different topics. The project has three types of subtasks: reliability analysis method development, issues in level 2 and 3 PRA and cross-cutting subjects.

The areas in method development are related to human reliability analysis and reliability analysis of passive systems. In the human reliability analysis area the main focus has been on participation in the Nordic-German co-operation project EXAM-HRA. Another topic within the HRA was independent method development, which was to improve the Enhanced Bayesian THERP method developed at VTT.

In level 2 and 3 PRA the development efforts are directed at creating a greater understanding of dynamic reliability analysis methods for use in level 2 PRA and increasing the capability to perform level 3 PRA.

PRA crosscutting subjects are topics that involve a new approach to PRA or are about more general PRA application. The research subtopics for PRA cross-cutting subjects are imprecise probabilities in PRA and risk communication.

Specific goals in 2011

Specific goals in 2011 include participation in the EXAM-HRA co-operation project Phase 2, where human reliability analysis practices are compared among German, Swedish and Finnish nuclear power plants. Another goal for HRA research was to develop further the Enhanced Bayesian THERP method to allow improved modelling of context of the human actions with performance shaping factors.

In subtask of passive systems reliability analysis a case study was performed of the passive containment cooling system, analysed with the MELCOR simulation tool. The passive containment cooling system can be seen in Figure 2.8.3.1 and an example of the MELCOR simulation results can be seen in Figure 2.8.3.2. Additionally a framework for reliability analysis of passive systems was planned and a literature study was performed.

The goal in dynamic PRA subtask was to conduct method development in co-operation with KTH (Kungliga Tekniska Högskolan) and Scandpower of Sweden. However, the co-operation goals were not possible due to Swedish partners being unable to participate due to their lack of favourable funding decisions. Dynamic PRA topic was instead executed by organizing an international workshop at VTT on the topic of DPSA – Deterministic / Probabilistic Safety Assessment. Purpose of the workshop was to present different approaches to integration of the deterministic and probabilistic safety analyses, and propose a consensus research agenda for the future. Level 3 subtopic had no goals for 2011, since the plan was to start level 3 work in 2012.

For the PRA crosscutting subjects following goals were stated. For imprecise probabilities the goal for 2011 was to develop a methodology to integrate the imprecise probabilities approach into the usual PRA framework, and to develop an interpretation of the classical importance measures in the imprecise probabilities framework. Risk communication goal for 2011 was to develop a co-operation plan for an empirical research project with KTH Risk Philosophy department, and conduct a literature review into effective risk communication in a nuclear
power plant safety organization with the plant PRAs being the main source of risk information.

Figure 2.8.3.1. The passive containment cooling system analyzed as a pilot case in the passive systems reliability analysis subtask of PRADA project.

Figure 2.8.3.2. The pressure development in the drywell during simulations (MELCOR) with three heat exchangers. The dashed line is with no PCCS.
Deliverables in 2011

- Enhanced Bayesian THERP HRA method was further developed based on empirical evidence. The method was evaluated in comparison to the findings in EXAM-HRA project and the International HRA Study project. Improvement recommendations on the performance shaping factors based on the empirical sources were completed. Method development was reported in a VTT research report.

- Participation in the Nordic German co-operation EXAM-HRA Phase 2 was conducted in 2011. In Phase 2 work in 2011 plant visits were conducted to enhance the comparison of HRA scenario comparisons between the different plant PRAs. Additional case comparisons were conducted to identify plant design and operational differences, good HRA practices and strengths and weaknesses of the methods. Reported in the EXAM-HRA Phase summary report and PSAM11 paper.

- A pilot case study was performed on reliability of a passive system. Passive containment cooling system was analysed using MELCOR using the Olkiluoto 1 and 2 model. Reliability methods were applied but functionality of the PCCS was so low that no reliability figure was achieved. Also a literature review was performed on the current state of passive system reliability analysis methodologies.

- DPSA workshop was held at VTT in co-operation with KTH, Scandpower and Vattenfall. The proceedings were published as a VTT report. Another result of the DPSA workshop was the formation of an IDPSA network for parties interested in research of Integrated Deterministic / Probabilistic PSA topics.

- Imprecise probabilities subtask developed an approach where the point estimate probabilities in a standard NPP PRA model are treated as probability intervals. This approach allows enhanced ability to model uncertainties in PRAs. Additionally importance measures were developed to be used with imprecise probabilities. The findings were published in a conference article.

- In the risk communication subtask a literature review was performed and further work was planned for 2012 in co-operation with KTH.

2.9 Development of research infrastructure

In 2011 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 CFD grade 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, carrying out experiments with CFD grade
measurements in order to support computational methods and validation requires the adaption of new measuring devices and systems, PIV, WMS and process tomography among others.

Requirements for the data acquisition system when handling CFD grade data are higher due to the large amount of data to be transferred, handled, and finally stored. Upgrading the systems and ensuring the compatibility of the data acquisition used in different test facilities is essential to avoid maintaining of multiple software and system architectures.

To take the full advantage of introducing novel measuring techniques, also the traditional control and measuring infrastructure has to be up to date. Ensuring the functions of systems in different experiment platforms and facilities requires stand-by or substituting components. Long delivery times or even the discontinuance of important components may cause unexpected delays in research projects. Thus, the essential replacement components have to be available on-site. Moving the instruments back and forth between the test facilities is not effective.

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, has been used at LUT for this purpose. However, the support of STRESA has been suspended and the software used in the database is not compatible with the operating systems to be taken into use at LUT in near future. This is why new replacing software with similar functions as in STRESA is being developed.

Specific goals in 2011

Specific goals in 2011 included ordering and commissioning of the PIV system. In the procurement, as the price of the system exceeded the national threshold value, the general terms of public procurement in supply contracts had to be obeyed. This caused some delay, but the PIV system was commissioned in late 2011 and applied first in EXCOP project in December 2011. Applicability of other advanced devices was also studied and the requirements of the electronics of the wire mesh sensor (WMS) were defined. In the data storage and distribution system development the conversion program of the existing database was finished as well as the programming of the user features such as browsing, searching and data enquiries. Programming of the maintenance features continues in 2012. To ensure the operability of the test facilities components related to instruments and measurements and also equipment enhancing the working safety has been bought. The process control system used in the PACTEL and PPOOLEX facilities has been installed and coded in a new platform. License for operating the test facilities using the updated control system has been granted by Inspecta. The operating manual of the control system has been written.

Deliverables in 2011

- PIV system was purchased and commissioned. System was applied in PPOOLEX experiments. A master’s thesis has been written.

- WMS electronics was proposed to be purchased and simple sensors to be tested in a pool system in atmospheric pressure.

- Software for data storage and distribution is in internal testing phase.
Components related to instruments and measurements and also equipment enhancing the working safety has been bought. The process control system used in the PACTEL and PPOOLEX facilities has been installed and coded in a new platform.

Figure 2.9.1.1. The averaged velocity field of the strong outflow phase in the exit of the blowdown pipe in PPOOLEX experiment recorded with PIV system.

2.9.2 Renewal of hot cell infrastructure (REHOT)

Many critical issues concerning plant life management for operating nuclear power plants are related to materials. Present plans for concurrent lifetime extension and power upgrading will eventually increase the needs to investigate and solve problems related to components and structural integrity. Degradation related to ageing of structures and components are important to safety and ageing management requires activities related to operation, inspection, testing, examination, maintenance and surveillance. These activities need to focus on effects how reactor environments and operating conditions affect strength and integrity of components and structures. In order to realize these activities critical infrastructure, e.g., hot cells equipped with handling, metallographic, microscopic, mechanical testing and autoclaves facilities is a necessity.

The present hot cell infrastructure at VTT and main part of the facilities have been built in the 1970’s and are, thus not technically up to date and are not able to adequately fulfil all present research requirements. It should be noticed also that there has been changes in property ownership status and there are plans for renovation work of the current facility which cause some concern of losing or downsizing the present hot cell infrastructure at VTT if new facilities do not become available.
VTT has carried out the conceptual engineering design (tarvekartoitus) and the detailed engineering design phases (hankesuunnitelma) of the new building construction project during previous years. The purpose of the new building construction is to bring together all nuclear research areas at VTT, i.e., radiochemistry, nuclear waste, dosimetry, fusion materials, failure analysis and mechanical characterisation of structural materials. Altogether this means laboratories and office buildings for about 150 persons. According to original planning the estimated timescale for a new building is about four years which means that new facilities could be available at the earliest in 2015.

VTT has presented the background, plans for new building and a proposal for funding the hot cell infrastructure for all parties of concern, i.e., authorities and power companies. At present, beginning of year 2011, the building project is on hold waiting for final decisions on how to proceed.

Specific goals in 2011

The basic assumption of the REHOT project is that a new building dedicated for nuclear research will be constructed in the near future. The objectives of this REHOT-project is to follow up, support and find out new approach for the renewal of infrastructure, to further define hot cell design and investment costs and to ensure continuity in education and training of personnel working in the hot cells. These objectives will be achieved in following three subtasks:

Procurements of Hot Cell equipment’s

The goal of this subtask is to further define the procurement of the hot cell equipment’s. The research equipment’s will be identified based on capabilities to carry out experiments on following research areas: surveillance programmes on pressure vessel steels, reactor internals, environment assisted cracking, next generation nuclear reactors and microscopy. The required research equipment’s are mostly similar and serve all these research areas but some equipment’s are specific for certain applications. Specimen preparation is an important area which supports all the above mentioned research areas and greatly increases the capabilities to carry out demanding failure analysis of irradiated and contaminated nuclear components.

Manipulator station and training

Feasibility and functionality of all operations carried out by manipulators form an important basis for the hot cell design. An additional important design requirement is radiation safety in all operations when it concerns handling of active materials, e.g., material transport in/out hot cells, specimen preparation and handling inside hot cells, intermediate active material storage and handling of active waste. Training and education of the hot cell staff is a key element in attaining a fully functional research facility for active materials.

In the first phase procurement of manipulators and design of a manipulator test station where various hot cell operations can be simulated is realised. The manipulator test station will be used for training and defining operational limits of the manipulators in various applications. Training and defining give valuable information for hot cell design and equipment requirements.

Support for renewal of infrastructure

This subtask will provide all necessary background information with a specific goal to determine the means for renewal of the present hot cell infrastructure. Additionally, as the
detailed engineering design study on the new building was completed by the end of year 2010 this subtask will make it possible to further refine the existing design, e.g., layout, dimensions and detailed technical requirements specified by different research topics. Also a preliminary plan for downsizing of the present hot cell activities possibly at alternative siting with or without other nuclear research activities or in the worst case scenario even close down of the activities was prepared.

**Deliverables in 2011**

The basic function of the present Hot Cells at VTT is related to the structural safety of the present nuclear power plants. The original layout, radiation protection and research equipment’s were particularly designed to carry out the surveillance programmes of the pressure vessel steels. From the late 70’s the layout and equipment’s have been evolved and at present following Hot Cell activities have been identified, Table 2.9.2.1.

**Table 2.9.2.1. Present Hot Cell activities, functions and equipment’s.**

<table>
<thead>
<tr>
<th>Activity</th>
<th>Function</th>
<th>Equipment’s</th>
<th>Notes</th>
</tr>
</thead>
<tbody>
<tr>
<td>Preparation of test specimens</td>
<td>Various geometries and sizes for mechanical tests and failure analysis</td>
<td>• EDM, Fanuc W0, 150mm&lt;br&gt;• EB CVE2000, 4kW&lt;br&gt;• Turning, milling, drilling, grinding machines&lt;br&gt;• MTS810, 100kN, servo hydraulic&lt;br&gt;• Instron5869, 50kN, electromechanical.&lt;br&gt;• Impact hammers&lt;br&gt;• PSW/MFL300, 300J&lt;br&gt;• PW5, 50J&lt;br&gt;• Pre-fatigue devices, laser&lt;br&gt;• Hardness tester, Duramin A300</td>
<td>3*&lt;br&gt;1&lt;br&gt;1.2</td>
</tr>
<tr>
<td>Mechanical testing</td>
<td>Various mechanical tests, e.g., tensile, fracture mechanism, impact, fatigue</td>
<td>• Diamond saw&lt;br&gt;• Struers Tenpol5&lt;br&gt;• Struers Planopol2&lt;br&gt;• Labopress3&lt;br&gt;• Ectching, ultrasound washer</td>
<td>3&lt;br&gt;1&lt;br&gt;3&lt;br&gt;1</td>
</tr>
<tr>
<td>Metallography</td>
<td>Specimen for optical and electron microscopy</td>
<td>• OM, Zeiss EZ3&lt;br&gt;• OM, LeitzWetzlar, Epiwert&lt;br&gt;• SEM Jeol JSM-5400&lt;br&gt;• TEM, FEI XL30ESEM</td>
<td>1&lt;br&gt;3&lt;br&gt;3</td>
</tr>
<tr>
<td>Microscopy</td>
<td>Optical and electron microscopy</td>
<td>• Water loops PWR, BWR&lt;br&gt;• Loading devices&lt;br&gt;• electrochemical measurements</td>
<td>1&lt;br&gt;1&lt;br&gt;1</td>
</tr>
<tr>
<td>Environmental testing</td>
<td>Stress corrosion cracking and fracture mechanical tests</td>
<td>• lead cell</td>
<td>1</td>
</tr>
<tr>
<td>Specimen reservoir</td>
<td>Tested specimens if not sent to the customer</td>
<td></td>
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</tbody>
</table>

1*)transfer, 2) transfer if needed, 3) investment

The basic design of the new Hot Cell facility is based on the concept of working in the shielded lead cells with mechanical manipulators. The design of the new Hot Cell facility is also based on the concept that all the present activities will be transferred to the new facility. It has also taken in to account that space is needed for new activities and further development of existing activities. The design basis for the number of lead cells is 9+2.

Cross-checking, revising and fine-tuning of the proposal for the renewal of the infrastructure were carried out.

A survey of several alternative locations within about 60 km from Otaniemi area was carried out. Extensive enquiry resulted in total of 6 potential existing real estates which could have been converted into Hot Cell facility. However, preliminary cost estimates indicated that no significant cost savings were foreseen. Similar results without any significant cost savings
were obtained after downsizing the research activities or after renovating existing buildings in Otaniemi area.

In-depth discussions on financing and service principles of particularly the Hot Cell infrastructure were carried out between all partners, e.g., power companies, ministry and VTT management.

VTT board decision on 15th December 2011 was to invest on new infrastructure for nuclear safety studies. The present planning of the renewal of the infrastructure is based on a new building in Otaniemi area. The new building for nuclear safety studies will consists of three functional buildings, e.g., office, laboratory and Hot cell buildings as can be seen in Figure 2.9.2.1. The capacity of the office building is 150 persons covering following research areas, e.g., active material research, radiochemistry, nuclear waste disposal, dosimetry, nuclear applied research, fusion research and iodine laboratory.

![Figure 2.9.2.1. Schematic drawing of the new nuclear safety building.](image-url)
3. Financial and statistical information

The planned and realised volumes of the SAFIR2014 programme in 2011 were 9.639 M€ and 9.494 M€ and 64 and 70 person years, respectively. The major funding partners were VYR with 5.149 M€, VTT with 2.859 M€, Aalto University with 0.220 M€, Fortum with 0.186 M€, NKS with 0.179 M€, TVO with 0.094 M€, and other partners with 0.795 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.

![Figure 3.1. Planned and realised financing of the SAFIR2014 programme in 2011.](image)

![Figure 3.2. Planned and realised costs of the SAFIR2014 programme in 2011.](image)
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 2011.

**Figure 3.4.** Planned and realised volumes by research area in 2011.

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 245 publications in 2011. Nearly half of the publications were research institute reports. The number of publications varies a lot between projects, as can be seen in table 3.1. The average number of publications was 3.5 per person-year, and the average number of scientific publications was 0.4 per person-year.
Table 3.1. Publications in the SAFIR2014 projects in 2011.

<table>
<thead>
<tr>
<th>Project acronym</th>
<th>Volume (person years)</th>
<th>Scientific</th>
<th>Conference papers</th>
<th>Res.inst. reports</th>
<th>Others</th>
<th>Total</th>
</tr>
</thead>
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<td>MANSCU</td>
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<td>5</td>
<td>3</td>
<td>3</td>
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<td>SAFEX2014</td>
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<td>0</td>
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<td>3</td>
</tr>
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<td>CORSICA</td>
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<td>HACAS</td>
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<td>SARANA</td>
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<td>3</td>
<td>5</td>
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<td>SAREMAN</td>
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<td>0</td>
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</tr>
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<td>CRISTAL</td>
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<td>KOURA</td>
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<td>BEPUE</td>
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<td>ESA</td>
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<td>3</td>
<td>1</td>
<td>4</td>
</tr>
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<td>8</td>
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</tr>
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</tr>
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<td>6</td>
</tr>
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<td>4</td>
<td>0</td>
<td>4</td>
</tr>
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</tr>
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<td>5</td>
<td>1</td>
<td>10</td>
</tr>
<tr>
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<td>1</td>
<td>3</td>
<td>0</td>
<td>4</td>
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<td>MAKOMON</td>
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<td>3</td>
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<td>0</td>
<td>8</td>
</tr>
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<td>RAIPSYS</td>
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</tr>
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<td>2</td>
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<td>3</td>
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<td>SESA</td>
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<td>0</td>
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<td>PRADA</td>
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<td>6</td>
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</tr>
<tr>
<td>ELAINE</td>
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<td>2</td>
<td>1</td>
<td>3</td>
</tr>
<tr>
<td>REHOT</td>
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<td>0</td>
<td>0</td>
<td>1</td>
<td>0</td>
<td>1</td>
</tr>
<tr>
<td>ADMIRE</td>
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</tr>
<tr>
<td><strong>TOTAL</strong></td>
<td><strong>69.9</strong></td>
<td><strong>27</strong></td>
<td><strong>67</strong></td>
<td><strong>121</strong></td>
<td><strong>30</strong></td>
<td><strong>245</strong></td>
</tr>
</tbody>
</table>
Altogether 7 higher academic degrees were obtained in the research projects in 2011, two Licentiate’s degrees and five 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 2011.

<table>
<thead>
<tr>
<th>Project acronym</th>
<th>Doctor</th>
<th>Licentiate</th>
<th>Master</th>
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<td>SARANA</td>
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<tr>
<td>PALAMA</td>
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<td>TERMOSAN</td>
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<td>1</td>
<td></td>
</tr>
<tr>
<td>RAIPSYS</td>
<td></td>
<td>1</td>
<td></td>
</tr>
<tr>
<td>WAPA</td>
<td></td>
<td></td>
<td>1</td>
</tr>
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<td>IMPACT2014</td>
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<td>1</td>
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</tr>
<tr>
<td>EXWE</td>
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<td></td>
<td>1</td>
</tr>
</tbody>
</table>
4. Programme management

During the administrative period (January 2011 – March 2012) the SAFIR2014 steering group held six meetings. The eight scientific reference groups met six times and the reference group for the development of research infrastructure five times, including the evaluation meetings of project proposals for 2012. During 2011, more than twenty Ad Hoc groups were set up to support projects and to improve 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 2011, small groups were set up to draft the Operational management handbook of the SAFIR2014 programme, and to draft supplements for the Framework document especially to identify complementary research needs due to the Fukushima Dai-ichi accident.

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.

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 of the projects in 2011
Management of safety culture throughout the lifecycle of nuclear plants (MANSCU):

Scientific publications

Gotcheva, N., Watts, G. and Oedewald, P. An Evolutionary Approach to Developing Smart and Safe Organizational Systems. *International Journal of Organizational Analysis*


Conference papers


Research institute reports


Sustainable and future oriented expertise (SAFEX2014):

Others


Pahkin Krista: presentation in SAFIR2010 seminar (March 2011): "Supervisor's role in knowledge management and expertise development: Results from a four-year project Expert work in a safety-critical environment"


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

Conference papers


Research institute reports


**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**


**Research institute reports**


Kim Björkman, Ola Bäckström, Jan-Erik Holmberg, Use of IEC 61508 in Nuclear Applications Regarding Software Reliability — Pre-study, VTT-R-09293-11, 2011.

**Others**

Yvonne Adolfsson, Jan Erik Holmberg, Göran Hultqvist, Pavel Kudinov, Ilkka Männistö, DPSA - Deterministic/Probabilistic Safety Analysis workshop proceedings, October 3-5, 2011, Espoo.


Tero Tyrväinen, Master’s Thesis: Risk Importance Measures and Common Cause Failures in Dynamic Flowgraph Methodology.

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

**Conference papers**


**Others**


Tommila, T. & Laarni, J. Concept of operations (ConOps) in the design of nuclear power plant instrumentation & control systems. Working report, 51 p.


**Criticality Safety and Transport methods in reactor analysis (CRISTAL):**

**Scientific publications**


**Conference papers**


Research institute reports


Three-dimensional reactor analyses (KOURA):

Conference papers


Conference papers in preparation


Research institute reports

Hovi V. New PORFLO version with unstructured mesh and parallelization. VTT Research Report VTT-R-01143-12. 27p.


Rintala J. Selection of the method for pin power reconstruction. VTT Research Report VTT-R-01163-12


Räty H., User’s manual for hot channel analyses with the one-dimensional reactor dynamics code TRAB. VTT Research Report VTT-R-01166-12. 78p.


Development of a Finnish Monte Carlo Reactor Physics Code (KÄÄRME):

Scientific publications


Conference papers


Research institute reports

Leppänen, J. “Serpent progress report 2010” VTT-R-01362-11

Neutronics, nuclear fuel and burnup (NEPAL):

Scientific publications


Research institute reports


Others


Extensive fuel modelling (PALAMA):

Conference papers


Tulkki, V. Development of VTT ENIGMA creep models, Enlarged Halden Programme Group Meeting 2011, Sandefjord, Norway, Oct. 3-6, 2011.


Research institute reports


Syrjälähti E., Characterization of a representative VVER-440 fuel rod with the statistical ENIGMA, VTT Research Report VTT-R-08464-11, 18 p.


Others


Application of Best Estimate Plus Uncertainty Evaluation Method (BEPUE):

Research institute reports


Enhancement of Safety Evaluation tools (ESA):

Conference papers


Research institute reports

Hillberg S., Simulation of NOKO emergency condenser EU experiments with APROS, VTT Research Report VTT-R-06319-11, 27 p.

Hillberg S., PANDA steady state isolation condenser experiments with APROS, VTT Research Report VTT-R-01259-12, 21 p.


Kurki J., Simulation of ROCOM Test 2.1 with APROS and TRACE, VTT Research Report VTT-R-00417-12, 15 p.

Hillberg S., Silde A., Modelling of PANDA PCC heat exchanger test T1.1 with APROS, VTT Research Report VTT-R-02563-12

Huhtanen R., Simulation of PANDA containment cooler experiment ST4.1 with Fluent, model refinements, VTT-R-02500-12, 13 p.


Experimental studies on containment phenomena (EXCOP):

Research institute reports


Others


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

Research institute reports


Tomas Brockmann, Detached-eddy Simulation of T-junction with OpenFOAM, CFD/MECHA-17-2012, 45 p.

Giteshkumar Patel, Vesa Tanskanen, CFD Simulation of Air Discharge into the Suppression Pool with OpenFOAM, Lappeenranta University of Technology, Nuclear Safety Research Unit, Laboratory of Nuclear Engineering, NuFoam 1/2011, 21 p.


Juho Peltola, twoPhaseNuFoam v0.3, VTT Research Report VTT-R-01073-12, 47 p., 2012.

Juho Peltola, Timo Pättikangas, Simulation of subcooled nucleate wall boiling with twoPhaseNuFoam v0.3: DEBORA5 and DEBORA6, VTT Research Report VTT-R-01038-12, 27 p., 2012.

Conference presentations


Numerical modeling of condensation pool (NUMPOOL):

Research institute reports


Improvement of PACTEL Facility Simulation Environment (PACSIM):

Conference papers

Vihavainen, J., Riikonen, V., Puustinen, M., Kyrki-Rajamäki, R., Modeling of the PACTEL Facility and Simulation of a Small Break LOCA Experiment with the TRACE V5.0 Code, The 14th International Topical Meeting on Nuclear Reactor Thermalhydraulics, NURETH-14, Toronto, Ontario, Canada, September 25-30, 2011, CD publication, article nro. 463.
Research institute reports


PWR PACTEL experiments (PAX):

Scientific publications


Research institute reports


Others


CFD modelling of NPP horizontal and vertical steam generators (SGEN):

Research institute reports


Core debris coolability (COOLOCE):

Scientific publications


Conference papers

Research institute reports


Chemistry of Fission Products (FISKE):

Research institute reports


Thermal Hydraulics of Severe Accidents (TERMOSAN):

Scientific publications

Nieminen, A. Core-melt behaviour inside the reactor pressure vessel during the late-phase of a severe accident. Master’s Thesis, Aalto University School of Engineering.

Conference papers

Sevón, T. MELCOR Modeling of Iodine Experiments in the THAI Facility. MCAP meeting, 22–23 September 2011, Bethesda, Maryland, USA.

Research institute reports


Transport and chemistry of fission products (TRAFI):

Conference papers


Research institute reports


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

Conference papers


Research institute reports


Karlsen, W. The role of non-monotonic loading in EAC- a literature review. VTT Research Report VTT-R-00867-12


Others


Fracture assessment of reactor circuit (FAR):

Conference papers

Research institute reports


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

Conference papers


Research institute reports


Jäppinen, T. Eddy Current Inspections to Detect Magnetite Depositions on 16x1,5 mm 316Ti tubes – Work Report, VTT Research Report VTT-R-01263-12. 8 p


RI-ISI Analyses and Inspection Reliability of Piping Systems (RAIPSYS):

Scientific publications


Cronvall, O. Structural lifetime, reliability and risk analysis approaches for power plant components and systems. VTT Publication P775. 264 p.

Research institute reports


Others


Advanced surveillance-techniques and embrittlement modelling (SURVIVE):

Conference papers


Research institute reports

Laukkonen, A., Library and wrapper for microstructures of polycrystalline and composite materials, VTT-R-00214-12, 18 p.

Water chemistry and plant operating reliability (WAPA):

Scientific publications


Research institute reports


Others

Väisänen, S., Magnettiitin zeta-potentiaali painevesivoimalaitoksen sekundääripiirillä vastaavissa olosuhteissa (Zeta Potential of Magnetite under PWR Secondary Circuit Conditions), Tampere University of Technology, 73 p. + 12 appendix pages, 2012. (Master of Science thesis)

Impact 2014 (IMPACT 2014):

Conference papers


Research institute reports


Aging Management of Concrete Structures in Nuclear Power Plants (MANAGE):

Conference papers


Research institute reports


Structural Mechanics Analysis of Soft and Hard Impacts (SMASH):

Scientific publications


Conference papers


Saarenheimo A., Tuomala M., Välikangas P. and Hakola I. IRIS_2010 Sensitivity studies on bending wall. SMiRT 21, 6-11 November, 2011, New Delhi, India

Tuomala M., Saarenheimo A., Calonius, K., Välikangas P. and Vepsä, A.. IRIS_2010 Sensitivity studies on punching wall. SMiRT 21, 6-11 November, 2011, New Delhi, India

Research institute reports


**Safety of Nuclear Power Plants (SESA):**

**Research institute reports**


**Extreme weather and nuclear power plants (EXWE):**

**Scientific publications**


**Conference papers**


**Research institute reports**


**Others**


Risk Assessment of Large Fire Loads (LARGO):

Scientific publications


Conference papers


Hostikka, S. & Matala, A. PYROLYSIS MODELING OF PVC CABLE MATERIALS. 21st International Conference on Structural Mechanics in Reactor Technology (SMiRT 21) -12th International Pre-Conference Seminar on “FIRE SAFETY IN NUCLEAR POWER PLANTS AND INSTALLATIONS”, 13-14.9.2011, München

Research institute reports


PRA Development and Application (PRADA):

Conference papers


**Research institute reports**


Silvonen, T., Reliability analysis for passive systems – A case study on a passive containment cooling system, VTT Research Report, VTT-R-06419-11, 38 p.


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

**Research institute reports**


Kotro, E., Pactel-laitteiston prosessiohjausjärjestelmän käyttöohje, ELAINE 2/2011, Lappeenranta University of Technology. (In Finnish)

**Others**


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

**Research institute reports**


**Administration of the research programme (ADMIRE):**

**Research institute reports**

Appendix 2

Participation in international projects and networks in 2011
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)

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)

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 Assessment)
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 WPNCS (Working Party on Nuclear Criticality Safety)
OECD/NEA EGUAM (Expert Group on Uncertainty Analysis in Modelling)
AER WG E (Atomic Energy Research, working group E: radwaste, spent fuel and decommissioning)

Three-dimensional reactor analyses (KOURA):
OECD/NEA NSC (Nuclear Science Committee)
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 Calculations and Uncertainty Analysis in Modelling)
Scientific Council of AER (Atomic Energy Research)
AER working group D on VVER safety analysis: participation in benchmark

Development of a Finnish Monte Carlo Reactor Physics Code (KÄÄRME):
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

Enhancement of Safety Evaluation tools (ESA):
OECD/NEA/CSNI Working Group on the Analysis and Management of Accidents (GAMA)
OECD/NEA Rig of Safety Assessment (ROSA-2)
OECD/NEA SESAR Thermal-hydraulics (SETH-2)
OECD/NEA Primary Coolant Loop Test Facility (PKL-2)
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)

Numerical modeling of condensation pool (NUMPOOL):
NKS project ENPOOL (Experimental and numerical studies on suppression pool issues)

PWR PACTEL experiments (PAX):
Preparations for OECD/PKL3 project (intended to address thermal-hydraulic safety issues for current PWR and new PWR design concepts through experiments at their integral test facility PKL, with some supporting tests done at Project members' facilities).

Core debris coolability (COOLOCE):
NKS project DECOSE (Debris Coolability and Steam Explosion)

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):
Phenbus 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-1 (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 II Programme (AeRosol Trapping In a Steam generaTor)
SARNET2 Network of Excellence / Source Term work package (WP8) (Euratom FP7)

Environmental influence on cracking susceptibility and ageing of nuclear materials (ENVIS):
CODAP (Component Operational Experience, Degradation and Ageing Programme)
OECD Halden Reactor Project
NULIFE Network of Excellence (Euratom FP6)
EPRI Alloy 690 expert group
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)
SE(T) Rond Robin (Recommended practice: fracture toughness testing using SE(T) samples with fixed-grip loading), co-ordinator CANMET

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):
ENIQ (European Network for Inspection and Qualification) Task Group Risk (TGR) activities
NULIFE Network of Excellence, IA-2-4 Expert Group Safety and reliability (Euratom FP6)

Advanced surveillance-techniques and embrittlement modelling (SURVIVE):
International Group for Radiation Damage Mechanisms (IGRDM)

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)

PRA Development and Application (PRADA):
OECD Halden Reactor Project: The International HRA Empirical Study
ASAMPSA2: Advanced Safety Assessment Methodologies: Level 2 PSA (Euratom FP7 project)
EXAM-HRA: Evaluation of Existing Applications and Guidance on Methods for HRA – NPSAG (Nordic PSA group)
DPSA Workshop: Deterministic / Probabilistic Safety Assessment Workshop – co-operation with Scandpower, KTH and Vattenfall

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

Academic degrees obtained in the projects in 2011
Safety evaluation and reliability analysis of nuclear automation (SARANA)

Master of Science in Technology:

Tero Tyrväinen, Risk Importance Measures and Common Cause Failures in Dynamic Flowgraph Methodology, Aalto University School of Science and Technology, 2011.

Extensive fuel modelling (PALAMA)

Licentiate in Technology:


Thermal hydraulics of severe accidents (TERMOSAN)

Master of Science in Technology:

Anna Nieminen, Core-melt behaviour inside the reactor pressure vessel during the late-phase of a severe accident, Aalto University School of Engineering, 2012.

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

Licentiate in technology

Otso Cronvall, Structural lifetime, reliability and risk analysis approaches for power plant components and systems. Aalto University School of Science and Technology, 2011.

Water chemistry and plant operating reliability (WAPA)

Master of Science in Technology:

Saija Väisänen, Model Checking Embedded Control Software, Tampere University of Technology, 2012.
Impact 2014 (IMPACT2014)

Master of Science in Technology:

Kalle Kaunisto, Contact force inversion from a linear system’s response, Aalto University School of Science and Technology, 2011.

Extreme weather and nuclear power plants (EXWE)

Master of Science:

Appendix 4

International travels in the projects in 2011
International travels in MANSCU project in 2011

Gotcheva, N. IAEA Consultancy meeting on ‘continual improvement of safety culture’. Wien, Austria. March 21-25, 2011

Oedewald, P.; Gotcheva, N. IAEA Consultancy meeting on ‘continual improvement of safety culture’. Wien, Austria. April 18-22, 2011


Oedewald, P. Nuclear technology 2011- Nordic symposium. Stockholm, Sweden December, 7-8, 2011

Project meetings:

Oedewald, P.; Macchi L. MOREMO kick off meeting. Halden, Norway. March 3-4, 2011

Oedewald, P.; Macchi L. MOREMO project meeting. Ringhals Varberg, Sweden. March 10-13, 2011

Macchi, L.; Reiman T., SADE project group meeting. Stockholm, Sweden. April 14, 2011

Oedewald, P.; Reiman, T. MOREMO project meeting. Oskarshamn, Sweden. May 2-3, 2011


Oedewald, P.; Macchi, L. MOREMO data collection. Ringhals Varberg, Sweden. July 4-6, 2011


International travels in SAFEX2014 project in 2011

None

International travels in CORSICA project in 2011

Nevalainen, R. CENELEC SC45AX meeting 5-6.12.2011, Bruxelles, Belgium.


**International travels in HACAS project in 2011**

Koskinen, H., EDF Outage Workshop, April, 1, 2011, Paris, France.


Laarni, J., HCI International 2011, July 9-14, 2011, Orlando, USA.

Laarni, J., Enlarged Halden Programme Group Meeting, October 2-7, 2011, Sandefjord, Norway.

Norros, L. GDR Psycho Ergo Overview and perspectives colloquy, April 4-6, 2011, Toulouse, France.

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


Norros, L., OECD/NEA WGHOF meeting and the NEA/IAEA workshop "Oversight and influencing of leadership and management for safety, including safety culture - Regulatory approaches and methods", September 26-28, Chester, England.

Savioja, P., EDF workshop, April 18, Paris, France.


**International travels in SARANA project in 2011**

Holmberg, Jan-Erik, Nordic PSA Group Meeting, February 2-3, 2012, Berlin, Germany

Holmberg, Jan-Erik, OECD/NEA Working Group RISK, Annual Meeting and DIGREL Task Group meeting, March 28-April 1, 2011, Paris, France

Holmberg, Jan-Erik, WGRSK DIGREL Task Group Meeting, May 16-19, 2011, Washington D.C., USA

Holmberg, Jan-Erik, Nordic PSA Conference – Castle Meeting 2011, 5–6 September 2011, Johannesberg Castle, Sweden

Holmberg, Jan-Erik, Janne Valkonen, Use of IEC 61508 as a basis for software analysis within nuclear PSA, 10th January, 2012, Stockholm, Sweden

Heljanko, Keijo, Computer Aided Verification Conference 2011, 12-22 July 2011, Snowbird, Utah, USA.

**International travels in SAREMAN project in 2011**


Uusitalo, E. Fourth International Workshop on Requirements Engineering and Law (RELAW 2011, in conjunction with the 19th IEEE International Requirements Engineering Conference). August 30th 2011, Trento, Italy.


**International travels in CRISTAL project in 2011**

Räty, A., SCALE training course, March 7-11, 2011, Paris, France

Rantanäki, K., First International Course Criticality Safety, March 28 - April 8, 2011, Saclay, France

Pusa, M., UAM benchmark meeting, April 13-15, 2011, Stockholm, Sweden

Häkkinen, S., AER Working Group E meeting, May 2nd - 4, 2011, Modra, Slovakia,

Serén, T., 14th International Symposium on Reactor Dosimetry (ISRD), May 22-27, 2011, Bretton Woods, New Hampshire, USA

Rantanäki, K., CMS Users Group Meeting, May 23rd -25, 2011, Stockholm, Sweden


**International travels in KOURA project in 2011**

Daavittila, A., Meeting of OECD/NEA Working Party on Scientific Issues of Reactor Systems (WPRS), February 2-4. 2011, Issy-les-Moulineaux, France

Syrjälähti E., Meeting of AER working group D and OECD/NEA Kalinin-3 benchmark workshop, April 12-13, Stockholm, Sweden
Räty H., Start-up meeting of the OECD/NEA Benchmark for the Oskarshamn-2 (O2) BWR Stability, April 13, Stockholm, Sweden

Daavittila, A., Meetings of OECD/NEA Nuclear Science Committee and the NEA Data Bank Executive Group, June 15-17, Issy-les-Moulineaux, France

Rintala J., FJOH summer school, August 24 – September 2, Karlsruhe, Germany

Rintala J., AER symposium, September 19-23, Dresden, Germany

**International travels in KÄÄRME project in 2011**


**International travels in NEPAL project in 2011**

None

**International travels in PALAMA project in 2011**


V. Tulkki, Halden VVER experiment Workshop, 11.5.2011 Budapest, Hungary.


V. Tulkki, FUMEX-III 3rd Research Coordination Meeting, 4.-8.12.2011, Vienna, Austria.
**International travels in BEPUE project in 2011**

Luukka J. 3D S.UN.COP seminar, March 28–April 8, 2011, Wilmington USA.

**International travels in ESA project in 2011**

Hänninen M., FONESYS code developer’s network meeting, January 1-February 5, 2011, Grenoble, France.

Hillberg S., TRACE/SNAP workshop, March 1-18, 2011, Maryland, USA.

Inkinen P., TRACE/SNAP workshop, March 1-18, 2011, Maryland, USA.


Kurki J., US-NRC Code Application and Maintenance (CAMP) meeting, November 5-10, 2011, Philadelphia, USA.

Karppinen I., OECD/PKL2 Programme Review Group meeting, November 7-9, 2011, Erlangen, Germany.

Silde A., EU SARNET WP7.3 meeting, November 23-24, 2011, Aachen, Germany.

Karppinen I., Northnet RoadMap 3 meeting, December 2, Västerås, Sweden.

Hänninen M., FONESYS code developer’s network meeting, December 11-15, 2011, Pisa Italy.


**International travels in EXCOP project in 2011**


International travels in NUFOAM project in 2011
None

International travels in NUMPOOL project in 2011

International travels in PACSIM project in 2011
None

International travels in PAX project in 2011
Heikki Purhonen, The 8th Meeting of the Programme Review Group and Management Board of the OECD PKL2 Project, AREVA NP, Erlangen, Germany, 8-9 November 2011.

International travels in SGEN project in 2011

International travels in COOLOCE project in 2011
None

International travels in FISKE project in 2011

International travels in TERMOSAN project in 2011
Sevón, T., CSARP (Cooperative Severe Accident Research Program) meeting & MCAP (MELCOR Cooperative Assessment Program) meeting. September 20–23, 2011, Bethesda, Maryland, USA.
**International travels in TRAFI project in 2011**

Auvinen, A., Kärkelä, T., Phebus FP, ISTP and SARNET follow-up meetings, 28.3-1.4.2011, Bergen, Netherlands.

Auvinen, A., Kärkelä, T., Phebus FP, ISTP and SARNET follow-up meetings, 17-20.10.2011, Aix-en-Provence, France.

Kärkelä, T., OECD/NEA STEM follow-up meeting (PRG + MB), 5.10.2011, Paris, France.

Auvinen, A., Kalilainen, J., ARTIST-2 PRG meeting, 24-25.1.2011, Villingen, Switzerland.


Kalilainen, J., Severe accident phenomenology course (SARNET2 network), 9-14.1.2011, Pisa, Italy.


**International travels in ENVIS project in 2011**


Ehrnstén, U. 1st CODAP project meeting, 18-19.5.2011, Paris, France.


International travels in FAR project in 2011

International travels in MAKOMON project in 2011

International travels in RAIPSYS project in 2011
Männistö, I., participation in ENIQ meeting in 17th of May in Manchester, UK, hosted by SERCO.

International travels in SURVIVE project in 2011
Valo, M., IGRDM-16, International Group on Radiation Damage Mechanisms, December 5-9, 2011, Santa Barbara, USA

International travels in WAPA project in 2011

International travels in IMPACT 2014 project in 2011
Calonius C., SMIRT 21. 6-11 November 2011. 21st International Conference on Structural Mechanics in Reactor Technology, India Habitat Centre, New Delhi, India.

International travels in MANAGE project in 2011


International travels in SMASH project in 2011

Sikanen, T., 12th International Pre-Conference Seminar on FIRE SAFETY IN NUCLEAR POWER PLANTS AND INSTALLATIONS. 13-14.9.2011, München, Germany.

Saarenheimo, A. SMiRT 21, 6-11 November, 2011, New Delhi, India.

International travels in SESA project in 2011

Fülöp, L., Visit to Paks NPP to map opportunities of cooperation for NPP related research. 30 August 2011, Hungary

International travels in EXWE project in 2011

Johansson, M., Baltic Sea Science Congress, August 22-26, 2011, St. Petersburg, Russia.


Jylhä, K., European Geosciences Union General Assembly 2011, 4.-8.4 2011, Austria


International travels in LARGO project in 2011

Hostikka, S., NBSG fire mote, 25.1.2011, Malmö, Sweden


Hostikka, S. 12th SMIRT Pre-Conference Seminar on Fire Safety in Nuclear Power Plants and Installations Hosted by TÜV SÜD, München, Germany; September 13-15, 2011

Hostikka, S. OECD PRISME program review and management groups. 11.-14.4.2011, St Maximin la Ste Beaume – France.

International travels in PRADA project in 2011

Männistö, I., Hukki, K., PRADA project meeting with KTH and Scandpower, May 9. 2011, Stockholm, Sweden

Männistö, I., Holmberg, J-E., Slotssmöte, September 5-6., 2011, Roslagen, Sweden
Männistö, I., EXAM-HRA project meeting, September 19-21.2011, Brunsbüttel, Germany

Männistö, I., EXAM-HRA project meeting, January 10-11. 2012, Malmö, Sweden

International travels in ELAINE project in 2011

Purhonen, H., Visit to PSI for PIV co-operation and to familiarize with other advanced measuring techniques in thermal hydraulics. March 7-11, 2011, PSI, Villigen, Switzerland.

International travels in REHOT project in 2011

None

International travels in ADMIRE project in 2011

Simola, K., Consultative Committee Euratom-Fission, 28.6.2011, Brussels, Belgium

Simola, K., SAFIR2014 visit to SSM, 1.9.2011, Stockholm, Sweden

Simola, K., Consultative Committee Euratom-Fission, 14.11.2011, Brussels, Belgium

Simola, K., Informal meeting on Fission research activities under Horizon2020 and Consultative Committee Euratom-Fission, 26.-27.1.2012, Brussels, Belgium
Appendix 5

The steering group, the reference groups and the scientific staff of the projects in 2011
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>
</tr>
<tr>
<td>Pentti Kauppinen</td>
<td>Technical Research Centre of Finland (VTT)</td>
</tr>
<tr>
<td>Liisa Heikinheimo</td>
<td>Teollisuuden Voima Oyj (TVO)</td>
</tr>
<tr>
<td>Pekka Pyy</td>
<td>Teollisuuden Voima Oyj (TVO)</td>
</tr>
<tr>
<td>Sami Hautakangas</td>
<td>Fortum Power and Heat Oy (Fortum)</td>
</tr>
<tr>
<td>Elizaveta Vainonen-Ahlgren</td>
<td>Fortum Power and Heat Oy (Fortum)</td>
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<tr>
<td>Juhani Hyvärinen</td>
<td>Fennovoima Oy</td>
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<tr>
<td>Nina Koivula (-10/2011)</td>
<td>Fennovoima Oy</td>
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<tr>
<td>Nina Lahtinen (10/2011-)</td>
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<tr>
<td>Rainer Salomaa</td>
<td>Aalto University</td>
</tr>
<tr>
<td>Sanna Syri</td>
<td>Aalto University</td>
</tr>
<tr>
<td>Riitta Kyrki-Rajamäki</td>
<td>Lappeenranta University of Technology (LUT)</td>
</tr>
<tr>
<td>Heikki Purhonen</td>
<td>Lappeenranta University of Technology (LUT)</td>
</tr>
<tr>
<td>Anneli Leppänen</td>
<td>Finnish Institute of Occupational Health (TTL)</td>
</tr>
<tr>
<td>Jaana Avolahti</td>
<td>Ministry of Employment and the Economy (TEM)</td>
</tr>
<tr>
<td>Lars Skånberg (11/2011-)</td>
<td>Swedish Radiation Safety Authority (SSM)</td>
</tr>
<tr>
<td>Jorma Aurela, TEM &amp; VYR contact person</td>
<td>Ministry of Employment and the Economy (TEM)</td>
</tr>
<tr>
<td>Harri Heimbürger, Expert</td>
<td>Radiation and Nuclear Safety Authority (STUK)</td>
</tr>
<tr>
<td>Kaisa Simola, Director of SAFIR2014, Secretary of the Steering Group</td>
<td>Technical Research Centre of Finland (VTT)</td>
</tr>
</tbody>
</table>
SAFIR2014 Reference Groups

1. Man, organisation and society

<table>
<thead>
<tr>
<th>Person</th>
<th>Organisation</th>
</tr>
</thead>
<tbody>
<tr>
<td>Matti Vartiainen, chairperson</td>
<td>Aalto</td>
</tr>
<tr>
<td>Kirsi Levä</td>
<td>STUK</td>
</tr>
<tr>
<td>Milka Holopainen (-11/2011)</td>
<td>STUK</td>
</tr>
<tr>
<td>Ann-Mari Sunabacka-Starck (11/2011-)</td>
<td>STUK</td>
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<tr>
<td>Leena Norros</td>
<td>VTT</td>
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<tr>
<td>Heli Talja</td>
<td>VTT</td>
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<tr>
<td>Petri Koistinen</td>
<td>TVO</td>
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<tr>
<td>Jari Tauluvuori</td>
<td>TVO</td>
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<tr>
<td>Tellervo Brandt</td>
<td>Fortum</td>
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<tr>
<td>Magnus Halin</td>
<td>Fortum</td>
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<tr>
<td>Matti Kattainen</td>
<td>Fortum</td>
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<tr>
<td>Tiina Tigerstedt</td>
<td>Fennovoima</td>
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<tr>
<td>Mikko Merikari</td>
<td>Fennovoima</td>
</tr>
<tr>
<td>Anna-Maria Teperi</td>
<td>(HF expert)</td>
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<tr>
<td>Per-Olof Sandén (11/2011-)</td>
<td>SSM</td>
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</table>

2. Automation and control room

<table>
<thead>
<tr>
<th>Person</th>
<th>Organisation</th>
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<tbody>
<tr>
<td>Olli Hoikkala, chairperson</td>
<td>TVO</td>
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<tr>
<td>Mika Koskela</td>
<td>STUK</td>
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<tr>
<td>Heimo Takala</td>
<td>STUK</td>
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<tr>
<td>Minna Tuomainen</td>
<td>STUK</td>
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<tr>
<td>Anna Aspelund (-11/2011)</td>
<td>STUK</td>
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<tr>
<td>Jari Hämäläinen</td>
<td>VTT</td>
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<td>Olli Ventä</td>
<td>VTT</td>
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<tr>
<td>Mauri Viitasalo</td>
<td>TVO</td>
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<tr>
<td>Martti Välisuo</td>
<td>Fortum</td>
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<tr>
<td>Ville Nurnilaukas</td>
<td>Fortum</td>
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<td>Juha Sirola</td>
<td>Fennovoima</td>
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<tr>
<td>Ilkka Niemelä</td>
<td>Aalto</td>
</tr>
<tr>
<td>Yvonne Johansson (11/2011-)</td>
<td>SSM</td>
</tr>
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</table>
3. Fuel research and reactor analysis

<table>
<thead>
<tr>
<th>Person</th>
<th>Organisation</th>
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<tbody>
<tr>
<td><strong>Riku Mattila, chairperson</strong></td>
<td>STUK</td>
</tr>
<tr>
<td>Risto Sairanen</td>
<td>STUK</td>
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<tr>
<td>Lena Hansson-Lyyra (10/2011-)</td>
<td>STUK</td>
</tr>
<tr>
<td>Markku Hänninen</td>
<td>VTT</td>
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<tr>
<td>Seppo Tähtinen</td>
<td>VTT</td>
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<tr>
<td>Petri Kotiluoto</td>
<td>VTT</td>
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<tr>
<td>Kari Ranta-Puska</td>
<td>TVO</td>
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<tr>
<td>Mikael Solala</td>
<td>TVO</td>
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<tr>
<td>Laura Kekkonen</td>
<td>Fortum</td>
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<tr>
<td>Tuukka Lahtinen</td>
<td>Fortum</td>
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<tr>
<td>Malla Seppälä</td>
<td>Fennovoima</td>
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<tr>
<td>Pertti Aarnio</td>
<td>Aalto</td>
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</table>

4. Thermal hydraulics

<table>
<thead>
<tr>
<th>Person</th>
<th>Organisation</th>
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<tbody>
<tr>
<td><strong>Eero Virtanen, chairperson</strong></td>
<td>STUK</td>
</tr>
<tr>
<td>Nina Lahtinen (-10/2011)</td>
<td>STUK</td>
</tr>
<tr>
<td>Miikka Lehtinen (11/2011-)</td>
<td>STUK</td>
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<tr>
<td>Mikko Ilvonen</td>
<td>VTT</td>
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<tr>
<td>Anitta Hämäläinen</td>
<td>VTT</td>
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<tr>
<td>Juha Poikolainen</td>
<td>TVO</td>
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<td>Mikko Lemmetty</td>
<td>TVO</td>
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<td>Timo Toppila</td>
<td>Fortum</td>
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<td>Heikki Kantee</td>
<td>Fortum</td>
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<td>Nhan Huynh</td>
<td>Fennovoima</td>
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<td>Malla Seppälä</td>
<td>Fennovoima</td>
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<tr>
<td>Timo Siikonen</td>
<td>Aalto</td>
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<tr>
<td>Anne Jordan</td>
<td>LUT</td>
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<tr>
<td>Annelie Bergman (11/2011-)</td>
<td>SSM</td>
</tr>
<tr>
<td>Pekka Nurmiilaukas, Asiantuntijäsen</td>
<td>Platom</td>
</tr>
</tbody>
</table>
5. **Severe accidents**

<table>
<thead>
<tr>
<th>Person</th>
<th>Organisation</th>
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<tbody>
<tr>
<td><strong>Risto Sairanen, chairperson</strong></td>
<td>STUK</td>
</tr>
<tr>
<td>Lauri Pöllänen</td>
<td>STUK</td>
</tr>
<tr>
<td>Tomi Routamo (6/2011-)</td>
<td>STUK</td>
</tr>
<tr>
<td>Arja Saarenheimo</td>
<td>VTT</td>
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<tr>
<td>Ilona Lindholm</td>
<td>VTT</td>
</tr>
<tr>
<td>Kristiina Rusanen (-6/2011)</td>
<td>TVO</td>
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<tr>
<td>Antti Tarkiainen (6/2011-)</td>
<td>TVO</td>
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<tr>
<td>Janne Vahero</td>
<td>TVO</td>
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<tr>
<td>Tommi Purho (-11/2011)</td>
<td>Fortum</td>
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<tr>
<td>Marko Marjamäki (11/2011-)</td>
<td>Fortum</td>
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<tr>
<td>Mika Harti</td>
<td>Fortum</td>
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<tr>
<td>Nici Bergroth</td>
<td>Fennovoima</td>
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<tr>
<td>Jarmo Ala-Heikkilä</td>
<td>Aalto</td>
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<tr>
<td>Heikki Suikkanen</td>
<td>LUT</td>
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6. **Structural safety of reactor circuits**

<table>
<thead>
<tr>
<th>Person</th>
<th>Organisation</th>
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<tbody>
<tr>
<td><strong>Martti Vilpas, chairperson</strong></td>
<td>STUK</td>
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<td>Mika Bäckström</td>
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<td>Wade Karlsen</td>
<td>VTT</td>
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<tr>
<td>Arja Saarenheimo</td>
<td>VTT</td>
</tr>
<tr>
<td>Eila Lehmus</td>
<td>VTT</td>
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<tr>
<td>Erkki Muttilainen</td>
<td>TVO</td>
</tr>
<tr>
<td>Anneli Reinvall (-10/2011)</td>
<td>TVO</td>
</tr>
<tr>
<td>Kimmo Tompuri (10/2011-)</td>
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<tr>
<td>Antti Kallio</td>
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<tr>
<td>Alpo Neuvonen</td>
<td>Fortum</td>
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<tr>
<td>Petri Kytöölä</td>
<td>Fortum</td>
</tr>
<tr>
<td>Ossi Hietanen</td>
<td>Fortum</td>
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<tr>
<td>Wolfgang Mayinger</td>
<td>Fennovoima</td>
</tr>
<tr>
<td>Laura Rissanen</td>
<td>Fennovoima</td>
</tr>
<tr>
<td>Gary Marquis</td>
<td>Aalto</td>
</tr>
<tr>
<td>Timo Merisaari</td>
<td>LUT</td>
</tr>
<tr>
<td>Kim Wallin</td>
<td>Suomen Akatemia</td>
</tr>
<tr>
<td>Peter Ekström (11/2011-)</td>
<td>SSM</td>
</tr>
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7. **Construction safety**

<table>
<thead>
<tr>
<th>Person</th>
<th>Organisation</th>
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</thead>
<tbody>
<tr>
<td>Pekka Välikangas, chairperson</td>
<td>STUK</td>
</tr>
<tr>
<td>Jukka Myllymäki</td>
<td>STUK</td>
</tr>
<tr>
<td>Heli Koukkari</td>
<td>VTT</td>
</tr>
<tr>
<td>Eila Lehmus</td>
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<tr>
<td>Vesa Hiltunen</td>
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<tr>
<td>Timo Kukkola</td>
<td>TVO</td>
</tr>
<tr>
<td>Joonas Koskinen</td>
<td>Fortum</td>
</tr>
<tr>
<td>Tapani Kukkola</td>
<td>Fortum</td>
</tr>
<tr>
<td>Aki Mattila</td>
<td>Fortum</td>
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<tr>
<td>Juha Matikainen</td>
<td>Fennovoima</td>
</tr>
<tr>
<td>Jari Puttonen</td>
<td>Aalto</td>
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8. **Probabilistic risk analysis (PRA)**

<table>
<thead>
<tr>
<th>Person</th>
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<tbody>
<tr>
<td>Reino Virolainen, chairperson</td>
<td>STUK</td>
</tr>
<tr>
<td>Jouko Marttila</td>
<td>STUK</td>
</tr>
<tr>
<td>Ilkka Niemelä</td>
<td>STUK</td>
</tr>
<tr>
<td>Esko Mikkola</td>
<td>VTT</td>
</tr>
<tr>
<td>Irina Aho-Mantila</td>
<td>VTT</td>
</tr>
<tr>
<td>Ilona Lindholm</td>
<td>VTT</td>
</tr>
<tr>
<td>Jan-Erik Holmberg</td>
<td>VTT</td>
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<tr>
<td>Risto Himanen</td>
<td>TVO</td>
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<tr>
<td>Jari Pesonen</td>
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<tr>
<td>Kalle Jänkälä</td>
<td>Fortum</td>
</tr>
<tr>
<td>Toivo Kivirinta</td>
<td>Fortum</td>
</tr>
<tr>
<td>Nici Bergroth</td>
<td>Fennovoima</td>
</tr>
<tr>
<td>Ahti Salo</td>
<td>Aalto</td>
</tr>
<tr>
<td>Tomas Jelinek (11/2011-)</td>
<td>SSM</td>
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### 9. Development of research infrastructure

<table>
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<tr>
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<th>Organisation</th>
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<tbody>
<tr>
<td><strong>Timo Vanttola, chairperson</strong></td>
<td>VTT</td>
</tr>
<tr>
<td>Martti Vilpas</td>
<td>STUK</td>
</tr>
<tr>
<td>Keijo Valtonen</td>
<td>STUK</td>
</tr>
<tr>
<td>Marja-Leena Järvinen</td>
<td>STUK</td>
</tr>
<tr>
<td>Pentti Kauppinen</td>
<td>VTT</td>
</tr>
<tr>
<td>Liisa Heikinheimo</td>
<td>TVO</td>
</tr>
<tr>
<td>Esa Mannola</td>
<td>TVO</td>
</tr>
<tr>
<td>Sami Hautakangas</td>
<td>Fortum</td>
</tr>
<tr>
<td>Harri Tuomisto</td>
<td>Fortum</td>
</tr>
<tr>
<td>Juhani Hyvärinen</td>
<td>Fennovoima</td>
</tr>
<tr>
<td>Christoffer Ehrnrooth</td>
<td>Fennovoima</td>
</tr>
<tr>
<td>Pertti Aarnio</td>
<td>Aalto</td>
</tr>
<tr>
<td>Riitta Kyrki-Rajamäki</td>
<td>LUT</td>
</tr>
<tr>
<td>Heikki Purhonen</td>
<td>LUT</td>
</tr>
<tr>
<td>Jorma Aurela</td>
<td>TEM</td>
</tr>
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</table>
Project personnel

Project Management of safety culture throughout the lifecycle of nuclear plants (MANSCU)
Turvallisuuskulttuurin hallinta laitosten elinkaarhen 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; research tasks, especially FRAM in MOREMO</td>
</tr>
<tr>
<td>Teemu Reiman, PhD (Psych)</td>
<td>VTT</td>
<td>Researcher tasks in DESIGN and MOREMO; safety culture theories</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>
<td>VTT</td>
<td>Researcher tasks in DESIGN, Human factors engineering</td>
</tr>
<tr>
<td>Nadezhda Gotcheva PhD (industrial management)</td>
<td>VTT</td>
<td>Researcher task in EVENTS and especially MODEL UPDATE; safety culture in networks</td>
</tr>
</tbody>
</table>

Sustainable and future oriented expertise (SAFEX2014)
Kestävää ja kehittyvää tulevaisuuden osaamista

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

<table>
<thead>
<tr>
<th>Person</th>
<th>Org.</th>
<th>Tasks</th>
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</thead>
<tbody>
<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>
</tr>
<tr>
<td>Tiina Kalliomäki-Levanto, DrTech</td>
<td>TTL</td>
<td>Design of data collecting instruments, quantitative and qualitative data analysis</td>
</tr>
<tr>
<td>Tanja Kuronen-Mattila, LicTech</td>
<td>Aalto</td>
<td>Design of data collecting instruments, quantitative and qualitative data analysis</td>
</tr>
<tr>
<td>Paula Poukka, trainee</td>
<td>Aalto</td>
<td>Master’s Thesis worker on subject “HR’s support to supervisors’ human skills in nuclear power plants”</td>
</tr>
<tr>
<td>Eila Järvenpää, Prof</td>
<td>Aalto</td>
<td>Advisor</td>
</tr>
</tbody>
</table>
Coverage and rationality of the software I&C safety assurance (CORSICA)
Turvallisuuskriittisten I&C ohjelmistojen täsmällinen ja kattava varmistaminen

Research organisation: VTT and FiSMA ry.
Project manager: Hannu Harju, VTT

<table>
<thead>
<tr>
<th>Person</th>
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<tbody>
<tr>
<td>Hannu Harju, LicTech</td>
<td>VTT</td>
<td>Evaluations coverage and rationality</td>
</tr>
<tr>
<td>Jukka Ranta, LicTech</td>
<td>VTT</td>
<td>Novel technologies</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 S4N assessment method and process</td>
</tr>
<tr>
<td>Timo Varkoi, LicTech</td>
<td>FiSMA ry.</td>
<td>Development of S4N assessment method and process</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>
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</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>Jari Lappalainen, LicScTech</td>
<td>VTT</td>
<td>Simulator environment design</td>
</tr>
<tr>
<td>Ilona 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>Marja Liinasuo, PhD</td>
<td>VTT</td>
<td>Operational concept in accident management, Automation awareness</td>
</tr>
<tr>
<td>Leena Salo, MScTech</td>
<td>VTT</td>
<td>Procedure usage in accident management, Safety automation concept</td>
</tr>
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</table>
Safety evaluation and reliability analysis of nuclear automation (SARANA)
Ydinvoima-automaation turvallisuuden ja luotettavuuden arviointi

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

<table>
<thead>
<tr>
<th>Person</th>
<th>Org.</th>
<th>Tasks</th>
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<tbody>
<tr>
<td>Janne Valkonen, MScTech</td>
<td>VTT</td>
<td>Project management, Applicability of different techniques and abstraction levels in modelling</td>
</tr>
<tr>
<td>Jan-Erik Holmberg, DrTech</td>
<td>VTT</td>
<td>Safety assessment of plant design concepts, Development of the reliability analysis tool YADRAT, Guidelines for reliability analysis of digital systems in PRA context, Use of IEC 61508 in nuclear applications regarding software reliability</td>
</tr>
<tr>
<td>Kim Björkman, MScTech</td>
<td>VTT</td>
<td>Guidelines for reliability analysis of digital systems in PRA context, Development of the reliability analysis tool YADRAT, Applicability of different techniques and abstraction levels in modelling</td>
</tr>
<tr>
<td>Jussi Lahtinen, MScTech</td>
<td>VTT</td>
<td>System level interfaces and timing issues, Formal analysis of large systems, Architecture-level modelling</td>
</tr>
<tr>
<td>Tero Tyrväinen, MScTech</td>
<td>VTT</td>
<td>Development of the reliability analysis tool YADRAT</td>
</tr>
<tr>
<td>Antti Pakonen, MScTech</td>
<td>VTT</td>
<td>Applicability of different techniques and abstraction levels in modelling</td>
</tr>
<tr>
<td>Keijo Heljanko, DrTech</td>
<td>Aalto</td>
<td>Probabilistic model checking, System level interfaces and timing issues, Formal analysis of large systems</td>
</tr>
<tr>
<td>Tuomas Launiainen, MScTech</td>
<td>Aalto</td>
<td>Formal analysis of large systems, System level interfaces and timing issues</td>
</tr>
<tr>
<td>Siert Wieringa, MScTech</td>
<td>Aalto</td>
<td>Formal analysis of large systems, System level interfaces and timing issues</td>
</tr>
<tr>
<td>Jonatan Ropponen, Trainee</td>
<td>Aalto</td>
<td>Probabilistic model checking</td>
</tr>
<tr>
<td>Janne Kauttio, Trainee</td>
<td>Aalto</td>
<td>System level interfaces and timing issues</td>
</tr>
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Safety requirements specification and management in nuclear power plants (SAREMAN)
Turvallisuusvaatimusten määrittely ja hallinta ydinvoimalaitoksilla

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

<table>
<thead>
<tr>
<th>Person</th>
<th>Org.</th>
<th>Tasks</th>
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</thead>
</table>
| Jari Laarni, PhD (psych.)       | VTT    | • Pre-study on the readability of safety-related documents in the nuclear field  
                                  |        | • The role of concept of operations in the design of nuclear power plant instrumentation & control systems |
| Teemu Tommila, M.Sc.(Eng.)      | VTT    | • Conceptual model for safety requirements specification and management  
                                  |        | • The role of concept of operations in the design of nuclear power plant instrumentation & control systems |
| Janne Valkonen, M.Sc.(Eng.)     | VTT    | • An example of system requirements specification for control rod control system upgrade  
                                  |        | • Conceptual model for safety requirements specification and management |
| Tomi Männistö, professor, D.Sc. (Tech.) | Aalto | Specifying requirements in structured natural language               |
| Mikko Raatikainen, M.Sc.(Tech)  | Aalto  | • Specifying requirements in structured natural language  
                                  |        | • Conceptual model for safety requirements specification and management |
| Eero Uusitalo, M.Sc.(Tech)      | Aalto  | Specifying requirements in structured natural language               |
Criticality safety and transport methods in reactor analysis (CRISTAL)
Kriittisyysturvallisuus ja kuljetusmenetelmät reaktorianalyysissä

Research organisation: VTT
Project manager: Karin Rantamäki, VTT

<table>
<thead>
<tr>
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<tbody>
<tr>
<td>Karin Rantamäki, DrTech</td>
<td>VTT</td>
<td>Project management, criticality safety, calculational methods</td>
</tr>
<tr>
<td>Petri Kotiluoto, PhD</td>
<td>VTT</td>
<td>calculational methods, nodal methods</td>
</tr>
<tr>
<td>Maria Pusa, MScTech</td>
<td>VTT</td>
<td>calculational methods</td>
</tr>
<tr>
<td>Antti Räty, MSc</td>
<td>VTT</td>
<td>calculational methods</td>
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<tr>
<td>Tuomas Viitanen, MScTech</td>
<td>VTT</td>
<td>calculational methods, criticality safety</td>
</tr>
<tr>
<td>Silja Häkkinen, DrTech</td>
<td>VTT</td>
<td>criticality safety</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</td>
</tr>
<tr>
<td>Auli Leveinen, assistant</td>
<td>VTT</td>
<td>Project management (assistant services)</td>
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Three-dimensional reactor analyses (KOURA)
Kolmiulotteiset reaktorianalyysit

Research organisation: VTT
Project manager: Elina Syrjälähti, VTT

<table>
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<tbody>
<tr>
<td>Syrjälähti Elina, MSc (Tech)</td>
<td>VTT</td>
</tr>
<tr>
<td>Daavittila Antti, MSc (Tech)</td>
<td>VTT</td>
</tr>
<tr>
<td>Hovi Ville, MSc (Tech)</td>
<td>VTT</td>
</tr>
<tr>
<td>Hämäläinen Anitta, DSc (Tech)</td>
<td>VTT</td>
</tr>
<tr>
<td>Ilvonen Mikko, LicSc (Tech)</td>
<td>VTT</td>
</tr>
<tr>
<td>Manninen Mikko, DSc (Tech)</td>
<td>VTT</td>
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<tr>
<td>Rintala Jukka, MSc (Tech)</td>
<td>VTT</td>
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<tr>
<td>Räty Hanna, MSc (Tech)</td>
<td>VTT</td>
</tr>
<tr>
<td>Seppälä Malla, MSc (Tech)</td>
<td>VTT (until 28.2.2011)</td>
</tr>
<tr>
<td>Taivassalo Veikko, PhLic (Phys)</td>
<td>VTT</td>
</tr>
<tr>
<td>Takasuo Eveliina, MSc (Tech)</td>
<td>VTT</td>
</tr>
<tr>
<td>Leveinen Auli, Assistant</td>
<td>VTT</td>
</tr>
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</table>
Development of a Finnish Monte Carlo reactor physics code (KÄÄRME)
Suomalaisen Monte Carlo-reactorifysiikkakoodin kehittäminen

Research organisation: VTT
Project manager: Jaakko Leppänen, VTT

<table>
<thead>
<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>
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Neutronics, nuclear fuel and burnup (NEPAL)
Neutroniikka, ydinpolttoaine ja palama

Research organisations: Aalto University
Project manager: Jarmo Ala-Heikkilä, Aalto

<table>
<thead>
<tr>
<th>Person</th>
<th>Org.</th>
<th>Tasks</th>
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<tbody>
<tr>
<td>Jarmo Ala-Heikkilä, D.Sc.(Tech.)</td>
<td>Aalto</td>
<td>Project manager</td>
</tr>
<tr>
<td>Seppo Sipilä, D.Sc.(Tech.)</td>
<td>Aalto</td>
<td>Deputy project manager</td>
</tr>
<tr>
<td>Aarno Isotalo, M.Sc.(Tech.)</td>
<td>Aalto</td>
<td>Computer simulations, analytical methods (military service VII/2011-)</td>
</tr>
<tr>
<td>Risto Vanhanen, M.Sc.(Tech.)</td>
<td>Aalto</td>
<td>Computer simulations, analytical methods (VI/2011-)</td>
</tr>
<tr>
<td>Ville Valtavirta, B.Sc.(Tech.)</td>
<td>Aalto</td>
<td>Computer simulations, analytical methods (VI/2011-)</td>
</tr>
<tr>
<td>Markus Ovaska, M.Sc.</td>
<td>Aalto</td>
<td>Computer simulations, analytical methods (X/2011-)</td>
</tr>
<tr>
<td>Pertti Aarnio, D.Sc.(Tech.)</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>
</tbody>
</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, DSc (Tech)</td>
<td>VTT</td>
<td>Investigation of uncertainty in system codes, implementation of boundary conditions enabling input from system code results in FRAPTRAN-GENFLO.</td>
</tr>
<tr>
<td>Anna Nieminen, MSc (Tech)</td>
<td>VTT</td>
<td>Review of phenomena relevant to the failed rods during the accidents.</td>
</tr>
<tr>
<td>Asko Arffman, MSc (Tech)</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, MSc (Tech)</td>
<td>VTT</td>
<td>Implementation of external statistical routine to ENIGMA, characterization of &quot;representative rod&quot; to be used in future simplified fuel model.</td>
</tr>
<tr>
<td>Jan-Olof Stengård, MSc (Tech)</td>
<td>VTT</td>
<td>Updating of statistical FRAPCON to the latest version, investigation of FRAPTRAN-GENFLO failure criteria.</td>
</tr>
<tr>
<td>Joonas Kättö, BSc (Tech)</td>
<td>VTT</td>
<td>Creation of validation database system for ENIGMA and the code SPACE for using it.</td>
</tr>
<tr>
<td>Seppo Kelppe, MSc (Tech)</td>
<td>VTT</td>
<td>Review of the PIE results of the Halden internal overpressure test series.</td>
</tr>
<tr>
<td>Tuomas Viitanen, MSc (Tech)</td>
<td>VTT</td>
<td>Coupling of ENIGMA and Serpent.</td>
</tr>
</tbody>
</table>

Application of best estimate plus uncertainty evaluation method (BEPUE)
Epävarmuusanalyysin soveltaminen best-estimate turvallisuusanalyysiiin

Research organisation: VTT
Project manager: Ismo Karppinen, VTT

<table>
<thead>
<tr>
<th>Person</th>
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<tbody>
<tr>
<td>Ismo Karppinen, MScTech</td>
<td>VTT</td>
<td>Project manager, identification of input parameters</td>
</tr>
<tr>
<td>Juha Luukka, MScTech</td>
<td>VTT</td>
<td>Deputy project manager, LBLOCA application</td>
</tr>
<tr>
<td>Elina Syrjälähti, MScTech</td>
<td>VTT</td>
<td>Identification of input parameters</td>
</tr>
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</table>
**Enhancement of safety evaluation tools (ESA)**

**Turvallisuusanalyysityökalujen kehittäminen**

Research organisation: VTT

Project manager: Ismo Karppinen, VTT

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<thead>
<tr>
<th>Person</th>
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<tbody>
<tr>
<td>Ismo Karppinen, MScTech</td>
<td>VTT</td>
<td>Project manager, participation in OECD/GAMA, follow-up of OECD/ROSA2, OECD/PKL2, co-ordination 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>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>Joona Kurki, LicTech</td>
<td>VTT</td>
<td>APROS 3D module testing and validation</td>
</tr>
<tr>
<td>Juha Kyttälä, MScTech</td>
<td>VTT</td>
<td>Verification of CFD calculation methods</td>
</tr>
<tr>
<td>Samps Lauerma, trainee</td>
<td>VTT</td>
<td>Validation of APROS</td>
</tr>
<tr>
<td>Juha Luukka, MScTech</td>
<td>VTT</td>
<td>Validation of APROS</td>
</tr>
<tr>
<td>Jarto Niemi, MScTech</td>
<td>VTT</td>
<td>Development of CFD models and error corrections in APROS 3D solver</td>
</tr>
<tr>
<td>Ari Silde, MScTech</td>
<td>VTT</td>
<td>Validation of APROS Containment</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>
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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 Räsänen, MScTech</td>
<td>LUT</td>
<td>Instrumentation, Data acquisition, Data conversion, Visualization, Control systems, Experiments</td>
</tr>
<tr>
<td>Vesa Tanskanen, MScTech</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 Pylkkö, 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, Research Trainee</td>
<td>LUT</td>
<td>Assessment of measurement techniques, Experiments</td>
</tr>
</tbody>
</table>

OpenFOAM CFD-solver for nuclear safety related flow simulations (NuFOAM)
OpenFOAM CFD -ratkaisija ydinturvallisuuden virtaussimulointeihin

Research organisation: VTT, Aalto, Fortum Power and Heat Oy, LUT
Project manager: Dr Timo Pättikangas, VTT

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<tr>
<th>Person</th>
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<tbody>
<tr>
<td>Timo Pättikangas, DTech</td>
<td>VTT</td>
<td>Project manager, two-phase CFD modeling</td>
</tr>
<tr>
<td>Juho Peltola, MScTech</td>
<td>VTT</td>
<td>Deputy project manager, two-phase CFD model development and validation</td>
</tr>
<tr>
<td>Jarto Niemi, MScTech</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, MScTech</td>
<td>Aalto</td>
<td>CFD modeling</td>
</tr>
<tr>
<td>Juhaveikko Ala-Juusela, MScTech</td>
<td>Aalto</td>
<td>CFD modeling</td>
</tr>
<tr>
<td>Karoliina Ekström, MScTech</td>
<td>Fortum</td>
<td>CFD modeling</td>
</tr>
<tr>
<td>Timo Toppila, MScTech</td>
<td>Fortum</td>
<td>CFD modeling</td>
</tr>
<tr>
<td>Vesa Tanskanen, DTech</td>
<td>LUT</td>
<td>CFD model development and validation</td>
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<tr>
<td>Giteshkumar Patel, MScTech</td>
<td>LUT</td>
<td>CFD model development and validation</td>
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**Numerical modeling of condensation pool (NUMPOOL)**

Lauhdutusaltaan numeerinen mallintaminen

Research organisation: VTT
Project manager: Timo Pättikangas, VTT

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<tr>
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<tbody>
<tr>
<td>Timo Pättikangas, DTech</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>
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**Improvement of PACTEL facility simulation environment (PACSIM)**

PACTEL koelaitteiston simulointiypäristön kehittäminen

Research organisation: LUT
Project manager: Juhani Vihavainen, LUT

<table>
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<tr>
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<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>
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</table>
## PWR PACTEL experiments (PAX)
PWR PACTEL kokeet

Research organisation: LUT  
Project manager: Vesa Riikonen, LUT

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<tr>
<th>Person</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>Juhani Vihavainen, Lic.Tech</td>
<td>LUT</td>
<td>TRACE code modeling and calculations</td>
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<tr>
<td>Antti Rantakaulio, MScTech</td>
<td>LUT</td>
<td>TRACE code modeling and calculations</td>
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<tr>
<td>Joonas Telkkä, Diploma worker</td>
<td>LUT</td>
<td>Studies to apply void fraction measuring methods</td>
</tr>
<tr>
<td>Antti Räsänen, 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>
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## Modelling of pressure transients in steam generators (SGEN)
Ydinvoimalaitosten vaaka- ja pystyhöyrystinten mallintaminen 3D virtauslaskennalla

Research organisation: VTT  
Project manager: Timo Pättikangas, VTT

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<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>
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<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>
</tbody>
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Core debris coolability (COOLOCE)
Partikkelimaisen sydänromun jäähdytettävyys

Research organisation: VTT
Project manager: Eveliina Takasuo, VTT

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<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, computational modelling</td>
</tr>
<tr>
<td>Tuomo Kinnunen, Engineer</td>
<td>VTT</td>
<td>COOLOCE experiments: design, installation and maintenance work</td>
</tr>
<tr>
<td>Pekka H. Pankakoski, MScTech</td>
<td>VTT</td>
<td>COOLOCE experiments: design, installation and reporting</td>
</tr>
<tr>
<td>Stefan Holmström, DrTech</td>
<td>VTT</td>
<td>COOLOCE experiments: design, reporting and management</td>
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<tr>
<td>Ville Hovi, MScTech</td>
<td>VTT</td>
<td>Computational modelling and PORFLO code development</td>
</tr>
<tr>
<td>Tuomo Sevón, MScTech</td>
<td>VTT</td>
<td>Deputy project manager</td>
</tr>
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</table>

Chemistry of fission products (FISKE)
Fissiotuotteiden kemia

Research organisation: VTT
Project manager: Tommi Kekki, VTT

<table>
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<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>Riitta Zilliacus, MSc</td>
<td>VTT</td>
<td>Deputy project manager, nitric acid formation</td>
</tr>
<tr>
<td>Lamminmäki Suvi, MSc Trainee</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, Technician</td>
<td>VTT</td>
<td>Nitric acid formation</td>
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<tr>
<td>Könönen Niina, MScTech</td>
<td>VTT</td>
<td>MELCOR calculations</td>
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<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>
</tr>
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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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<tr>
<th>Person</th>
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<tbody>
<tr>
<td>Tuomo Sevón, MScTech</td>
<td>VTT</td>
<td>PCC calculations with MELCOR, CSARP, OECD THAI-2, project management</td>
</tr>
<tr>
<td>Anna Nieminen, BScTech</td>
<td>VTT</td>
<td>Master’s thesis on melt pool behavior in reactor lower head</td>
</tr>
<tr>
<td>Ilona Lindholm, MScTech</td>
<td>VTT</td>
<td>OECD SERENA-2, instruction of the master’s thesis</td>
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### Transport and chemistry of fission products (TRAFI)

*Fissiotuotteiden kulkeutuminen ja kemia*

Research organisation: VTT  
Project manager: Teemu Kärkelä, VTT

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<tr>
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<tbody>
<tr>
<td>Teemu Kärkelä, MScTech</td>
<td>VTT</td>
<td>Iodine experiments, Participation in Phibus, ISTP and STEM projects</td>
</tr>
<tr>
<td>Ari Auvinen, MScTech</td>
<td>VTT</td>
<td>Participation in Phibus, ISTP, STEM and ARTIST2 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>
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<tr>
<td>Suvi Lamminmäki, Research</td>
<td>VTT</td>
<td>Radio tracer measurements - iodine experiments</td>
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<td>Trainee</td>
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<tr>
<td>Jaana Rantanen, Technician</td>
<td>VTT</td>
<td>Chemical analysis - iodine experiments</td>
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<tr>
<td>Tuula Kajolinna, Engineer</td>
<td>VTT</td>
<td>Gas compound 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, Participation in ARTIST2 project</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>
</tr>
</tbody>
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Environmental influence on cracking susceptibility and ageing on nuclear materials (ENVIS)
Ympäristön vaikutus ydinvoimalaitosmateriaalien murtumiseen ja vanhenemiseen

Research organisation: VTT
Project manager: Ulla Ehrnstén, VTT

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<tr>
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<tbody>
<tr>
<td>Ulla Ehrnstén, MScTech</td>
<td>VTT</td>
<td>Project manager, characterisation of materials, author of SOAR on thermal ageing, international co-operation, mentor of YG</td>
</tr>
<tr>
<td>Matias Ahonen, MScTech</td>
<td>VTT</td>
<td>Responsible for performing fracture toughness tests</td>
</tr>
<tr>
<td>Wade Karlsen, DrTech</td>
<td>VTT</td>
<td>Responsible for characterisation of irradiated stainless steels, international co-operation, mentor of YG, author of SOAR on fatigue mechanisms</td>
</tr>
<tr>
<td>Janne Pakarinen, DrTech</td>
<td>VTT</td>
<td>Characterisation of materials using TEM and SEM</td>
</tr>
<tr>
<td>Pasi Kuivalainen, Research Engineer</td>
<td>VTT</td>
<td>Responsible for autoclave testing and mentoring autoclave workers</td>
</tr>
<tr>
<td>Juha-Matti Autio, MScTech</td>
<td>VTT</td>
<td>Development of creep testing device, characterisation of mock-up</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>Seppo Tähtinen, MScTech</td>
<td>VTT</td>
<td>Responsible for SOAR on fuel clad materials</td>
</tr>
<tr>
<td>Mykola Ivanchenko, DrTech</td>
<td>Aalto</td>
<td>Responsible for fatigue testing in air</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>Hannu Hänninen, Prof.</td>
<td>Aalto</td>
<td>Scientific support</td>
</tr>
</tbody>
</table>
Fracture assessment of reactor circuit (FAR)
Reaktoripiirin murtumisriskin arviointi

Research organisation: VTT
Project manager: Päivi Karjalainen-Roikonen, VTT

<table>
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<tr>
<td>Päivi Karjalainen-Roikonen,</td>
<td>VTT</td>
<td>Low constraint, DMW, LBB</td>
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<tr>
<td>MScTech</td>
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<tr>
<td>Heikki Keinänen, MScTech</td>
<td>VTT</td>
<td>LBB, FEM methods</td>
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<tr>
<td>Juha Kuutti, MScTech</td>
<td>VTT</td>
<td>XFEM</td>
</tr>
<tr>
<td>Kalle Kaunisto, MScTech</td>
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<td>Tapio Planman, MScTech</td>
<td>VTT</td>
<td>WPS low constraint</td>
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<td>Lauri Elers, MScTech</td>
<td>VTT</td>
<td>low constraint</td>
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<tr>
<td>Pekka Nevasmaa, MScTech</td>
<td>VTT</td>
<td>DMW</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

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<tbody>
<tr>
<td>Tarja Jäppinen, Lic.Sc. (Tech.)</td>
<td>VTT</td>
<td>Project manager, Eddy current applications</td>
</tr>
<tr>
<td>Ari Koskinen, M.Sc. (Tech.)</td>
<td>VTT</td>
<td>Deputy project manager, Ultrasonic reflectors</td>
</tr>
<tr>
<td>Jonne Haapalainen, M.Sc.</td>
<td>VTT</td>
<td>Ultrasonic simulations and reflectors, Digital radiography</td>
</tr>
<tr>
<td>Esa Leskelä, M.Sc. (Tech.)</td>
<td>VTT</td>
<td>Ultrasonic simulations</td>
</tr>
<tr>
<td>Stefan Sandlin, M.Sc.</td>
<td>VTT</td>
<td>Nonlinear ultrasonics</td>
</tr>
<tr>
<td>Kari Lahdenperä, M.Sc. (Tech.)</td>
<td>VTT</td>
<td>Eddy current applications</td>
</tr>
<tr>
<td>Antti Tuhti, Engineer</td>
<td>VTT</td>
<td>Ultrasonic reflectors</td>
</tr>
<tr>
<td>Matti Sarkimo, LicTech</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

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<thead>
<tr>
<th>Person</th>
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<th>Tasks</th>
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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 rupture risks and POD sensitivity analyses</td>
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<tr>
<td>Ari Vepsä, MScTech</td>
<td>VTT</td>
<td>Deputy project manager, probabilistic simulations and sampling methods</td>
</tr>
</tbody>
</table>

Advanced surveillance-techniques and embrittlement modelling (SURVIVE)
Kehittynyt surveillance-teknikka ja materiaaliominais-suuksien luotettava mallintaminen

Research organisation: VTT
Project manager: Matti Valo, VTT

<table>
<thead>
<tr>
<th>Person</th>
<th>Org.</th>
<th>Tasks</th>
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</thead>
<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 organisations: VTT
Project manager: Timo Saario, VTT

<table>
<thead>
<tr>
<th>Person</th>
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<tbody>
<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>Seppo Peltonen, Research Assistant</td>
<td>VTT</td>
<td>Equipment design</td>
</tr>
<tr>
<td>Taru Lehtikuusi, Research Assistant</td>
<td>VTT</td>
<td>Chemistry control</td>
</tr>
<tr>
<td>Saija Väisänen, Research Trainee</td>
<td>VTT</td>
<td>Measurement of zeta-potential at PWR secondary side conditions</td>
</tr>
</tbody>
</table>

**Impact 2014 (IMPACT 2014)**

Research organisation: VTT
Project manager: Ilkka Hakola, VTT, Assistant project manager: Ari Vepsä

<table>
<thead>
<tr>
<th>Person</th>
<th>Title</th>
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<tbody>
<tr>
<td>Ilkka Hakola, MScTech</td>
<td>MScTech</td>
<td>VTT</td>
<td>Tasks: 1 Test apparatus; 2 Measurements, 3 Pre-calculations, 4 Testing, 5 Archiving</td>
</tr>
<tr>
<td>Vepsä Ari</td>
<td>MScTech</td>
<td>VTT</td>
<td>Assistant project manager 1 2 3 4 5</td>
</tr>
<tr>
<td>Calonius Kim Mathias</td>
<td>MScTech</td>
<td>VTT</td>
<td>2 4</td>
</tr>
<tr>
<td>Halonen Matti</td>
<td>MScTech</td>
<td>VTT</td>
<td>1 2 5</td>
</tr>
<tr>
<td>Hietalahti Jouni Aslak</td>
<td>Research eng.</td>
<td>VTT</td>
<td>1 2 4 5</td>
</tr>
<tr>
<td>Järvinen Erkki Olavi</td>
<td>MScTech</td>
<td>VTT</td>
<td>2 4 5</td>
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<tr>
<td>Kalle Kaunisto</td>
<td>MScTech</td>
<td>VTT</td>
<td>3 4</td>
</tr>
<tr>
<td>Kuutti Juha Ilmari</td>
<td>MScTech</td>
<td>VTT</td>
<td>2 3 4</td>
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<tr>
<td>Lehmus Eila</td>
<td>MScTech</td>
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<td>2</td>
</tr>
<tr>
<td>Mäkinen Jukka Olavi</td>
<td>Research eng.</td>
<td>VTT</td>
<td>1 2 4 5</td>
</tr>
<tr>
<td>Patalainen Mikko</td>
<td>MScTech</td>
<td>VTT</td>
<td>3 4 5</td>
</tr>
<tr>
<td>Saarenheimo Arja Kyllikki</td>
<td>LicScTech</td>
<td>VTT</td>
<td>2 3 4 5</td>
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<tr>
<td>Schlesier Erja Beata Hannele</td>
<td>Assistant</td>
<td>VTT</td>
<td>5</td>
</tr>
<tr>
<td>Sjöblom Ville</td>
<td>Research eng.</td>
<td>VTT</td>
<td>1 2 4 5</td>
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</table>

The material tests and casting have been done by VTT Expert Services Oy
Prestressing of concrete walls have been done by Tensicon consulting
Aging management of concrete structures in nuclear power plants (MANAGE)
Ydinvoimaloiden betonirakenteiden ikääntymisen hallinta

Research organisations: VTT, Aalto University
Project manager: Erkki Vesikari, VTT

<table>
<thead>
<tr>
<th>Person</th>
<th>Org.</th>
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<tbody>
<tr>
<td>Erkki Vesikari, LicTech</td>
<td>VTT</td>
<td>Planning of the platform, application and interfacing of the program ServiceMan, condition analyses based on samples.</td>
</tr>
<tr>
<td>Mikko Tuomisto, Engineer</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>Kim Calonius, MScTech</td>
<td>VTT</td>
<td>Planning of the platform, corrosion measurements in Loviisa 1 cooling water channels.</td>
</tr>
<tr>
<td>Esko Sistonen, DrTech</td>
<td>Aalto</td>
<td>Planning of the platform, Planning of the central database and the inspection database. Corrosion measurements in Loviisa 1 cooling water channels.</td>
</tr>
<tr>
<td>Fahim Al-Neshawy, LicTech</td>
<td>Aalto</td>
<td>Planning of the platform, Planning of the central database and the inspection database. Corrosion measurements in Loviisa 1 cooling water channels.</td>
</tr>
<tr>
<td>Jukka Piironen, MScTech</td>
<td>Aalto</td>
<td>Corrosion measurements in Loviisa 1 cooling water channels.</td>
</tr>
<tr>
<td>Olli-Pekka Kari, LicTech</td>
<td>Aalto</td>
<td>Modelling of degradation</td>
</tr>
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Structural mechanics analysis of soft and hard impacts (SMASH)
Rakenteiden mekaniikan menetelmiä pehmeiden ja kovien iskuuormitusten tarkasteluun

Research organisation: VTT
Project manager: Arja Saarenheimo, VTT

<table>
<thead>
<tr>
<th>Person</th>
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<tbody>
<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>
</tr>
<tr>
<td>Topi Sikanen, MScTech</td>
<td>VTT</td>
<td>Fire dynamic simulations</td>
</tr>
<tr>
<td>Risto Laukasti, LicTech</td>
<td>VTT</td>
<td>Explosion loads</td>
</tr>
<tr>
<td>Markku Tuomalanga, 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 organisation: VTT, Institute of Seismology (Uni. Of Helsinki), Aalto University, AF-Consult
Project manager: Ludovic Fülöp, VTT

<table>
<thead>
<tr>
<th>Person</th>
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<tbody>
<tr>
<td>Ludovic Fülöp, Ph.D.</td>
<td>VTT</td>
<td>Project manager; Coordination, Structural analysis – equipment qualification (SP1, SP2)</td>
</tr>
<tr>
<td>Saari Jouni, Ph.D.</td>
<td>AF</td>
<td>Deputy project manager; Seismic hazard.</td>
</tr>
<tr>
<td>Malm Marianne, M.Sc.</td>
<td>AF</td>
<td>Seismic hazard methods, earthquake data bank</td>
</tr>
<tr>
<td>Pekka Heikkinen, Prof.</td>
<td>SeI</td>
<td>Coordination of hazard studies (SP1)</td>
</tr>
<tr>
<td>Päivi Mäntyniemi, PhD,</td>
<td>SeI</td>
<td>Teaching - Earthquake hazard assessment</td>
</tr>
<tr>
<td>Ilmari Smedberg, trainee</td>
<td>SeI</td>
<td>Earthquake hazard assessment</td>
</tr>
<tr>
<td>Jari Puttonen, Prof.</td>
<td>Aalto</td>
<td>Overall planning of teaching activity</td>
</tr>
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Extreme weather and nuclear power plants (EXWE)
Sään ääri-ilmiöt ja ydinvoimalaitokset

Research organisation: FMI
Project manager: Milla Johansson, FMI

<table>
<thead>
<tr>
<th>Person</th>
<th>Org.</th>
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<tbody>
<tr>
<td>Milla Johansson, MSc</td>
<td>FMI</td>
<td>Project manager, sea level research</td>
</tr>
<tr>
<td>Kirsti Jylhä, PhD</td>
<td>FMI</td>
<td>Deputy project manager, research on heat waves</td>
</tr>
<tr>
<td>Kimmo Kahma, Prof</td>
<td>FMI</td>
<td>Sea level research</td>
</tr>
<tr>
<td>Hilkka Pellikka, MSc</td>
<td>FMI</td>
<td>Sea level research</td>
</tr>
<tr>
<td>Katri Leinonen</td>
<td>FMI</td>
<td>Sea level research</td>
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<tr>
<td>Kimmo Ruosteenoja, PhD</td>
<td>FMI</td>
<td>Research on heat waves</td>
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<tr>
<td>Seppo Saku, MSc</td>
<td>FMI</td>
<td>Research on heat waves</td>
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<tr>
<td>Hanna Tietäväinen, MSc</td>
<td>FMI</td>
<td>Research on heat waves</td>
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<tr>
<td>Hilppa Gregow, MSc</td>
<td>FMI</td>
<td>Research on extreme weather events</td>
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<tr>
<td>Matti Lahtinen</td>
<td>FMI</td>
<td>Research on heat waves</td>
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<tr>
<td>Jaakko Forsius</td>
<td>FMI</td>
<td>Research on heat waves</td>
</tr>
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Risk assessment of large fire loads (LARGO)
Suurten palokuormien riskien arviointi

Research organisation: VTT
Project manager: Simo Hostikka, VTT

<table>
<thead>
<tr>
<th>Person</th>
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<tbody>
<tr>
<td>Simo Hostikka, DSc (Tech)</td>
<td>VTT</td>
<td>Project manager, simulation model development, OECD PRISME</td>
</tr>
<tr>
<td>Terhi Kling, MSc (Tech)</td>
<td>VTT</td>
<td>Defence-in-depth, human reliability</td>
</tr>
<tr>
<td>Johan Mangs, PhD</td>
<td>VTT</td>
<td>Experiments on digital automation</td>
</tr>
<tr>
<td>Topi Sikanen, MSc (Tech)</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</td>
</tr>
<tr>
<td>Anna Matala, MSc (Tech)</td>
<td>VTT</td>
<td>Simulations of cable fires</td>
</tr>
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</table>

PRA development and application (PRADA)
Todennäköisyyspohjaisten riskianalyysien kehittäminen ja soveltaminen

Research organisation: VTT
Project manager: Ilkka Männistö, VTT

<table>
<thead>
<tr>
<th>Person</th>
<th>Org.</th>
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<tbody>
<tr>
<td>Ilkka Männistö, MScTech</td>
<td>VTT</td>
<td>Project Manager, EXAM-HRA participation, passive reliability analysis, risk communication</td>
</tr>
<tr>
<td>Jan-Erik Holmberg, DrTech</td>
<td>VTT</td>
<td>DPSA workshop technical committee, passive systems reliability planning</td>
</tr>
<tr>
<td>Ilkka Karanta, LicTech</td>
<td>VTT</td>
<td>Passive systems reliability analysis</td>
</tr>
<tr>
<td>Kristiina Hukki, M.A.</td>
<td>VTT</td>
<td>Human reliability analysis, risk communication</td>
</tr>
<tr>
<td>Taneli Silvonen, trainee</td>
<td>VTT</td>
<td>Passive systems reliability MELCOR modelling</td>
</tr>
<tr>
<td>Ilona Lindholm, MScTech</td>
<td>VTT</td>
<td>MELCOR modelling</td>
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Enhancement of Lappeenranta instrumentation of nuclear safety experiments (ELAINE)
Lappeenrannan ydinturvallisuuskoideen mitausten ajamukaistaminen

Research organisation: LUT
Project manager: Antti Räsänen, LUT

<table>
<thead>
<tr>
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<tbody>
<tr>
<td>Antti Räsänen, MScTech</td>
<td>LUT</td>
<td>Overall planning of the process control and data acquisition environment</td>
</tr>
<tr>
<td>Vesa Riikonen, MScTech</td>
<td>LUT</td>
<td>Data storage and distributing system development</td>
</tr>
<tr>
<td>Lauri Pyy, research trainee</td>
<td>LUT</td>
<td>Commissioning of the PIV system</td>
</tr>
<tr>
<td>Eetu Kotro, research trainee</td>
<td>LUT</td>
<td>Process control system</td>
</tr>
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Renewal of hot cell infrastructure (REHOT)
Kuumakammiotutkimusvalmiuksien uudistaminen

Research organisation: VTT
Project manager: Seppo Tähtinen, VTT

<table>
<thead>
<tr>
<th>Person</th>
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<tbody>
<tr>
<td>Seppo Tähtinen, MScTech</td>
<td>VTT</td>
<td>Hot Cells, mechanical tests</td>
</tr>
<tr>
<td>Tommi Kekki, MScTech</td>
<td>VTT</td>
<td>Other nuclear R&amp;D</td>
</tr>
<tr>
<td>Wade Karlsen, DrTech</td>
<td>VTT</td>
<td>Hot Cells</td>
</tr>
<tr>
<td>Timo Vanttola, DrTech</td>
<td>VTT</td>
<td>Project planning</td>
</tr>
<tr>
<td>Pentti Kauppinen, DrTech</td>
<td>VTT</td>
<td>Project planning</td>
</tr>
<tr>
<td>Seppo Vuori, DrTech</td>
<td>VTT</td>
<td>Project planning</td>
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Administration of the Research programme (ADMIRE)
Tutkimusohjelman hallinointi

Research organisation: VTT
Project manager: Kaisa Simola, VTT

<table>
<thead>
<tr>
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<tbody>
<tr>
<td>Kaisa Simola, DrTech</td>
<td>VTT</td>
<td>Project manager, programme administration, EU FP7 CCE-Fission Committee</td>
</tr>
<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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