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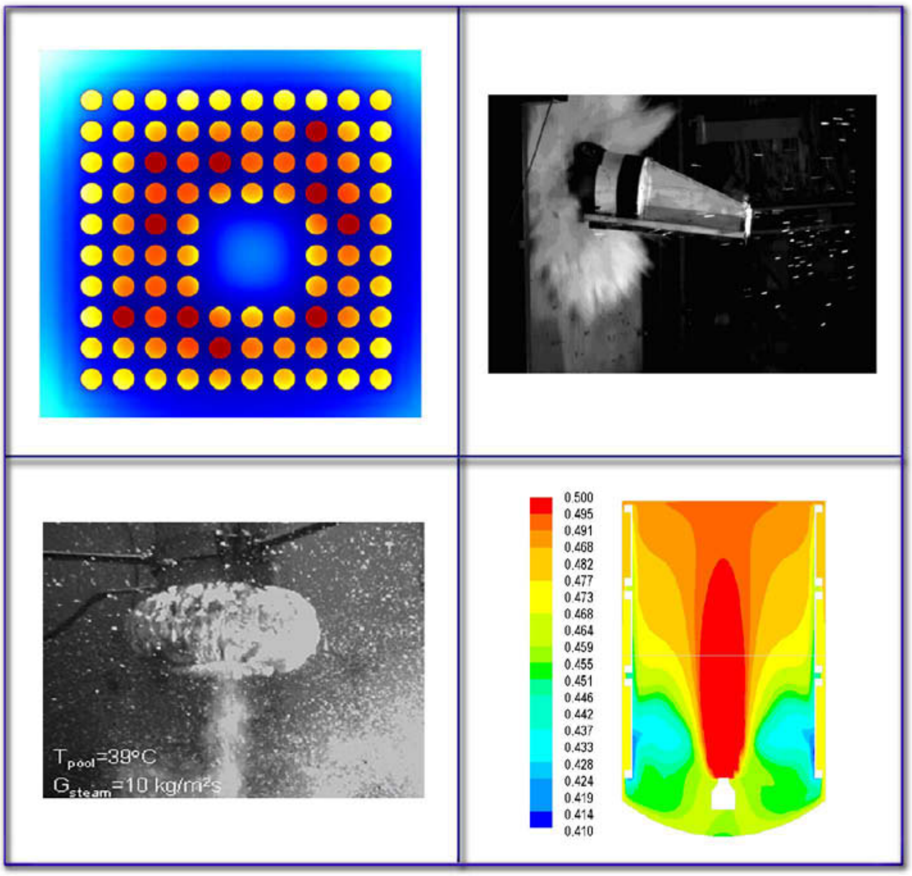

SAFIR The Finnish Research Programme on Nuclear Power Plant Safety 2003- 2006

Executive Summary

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The Finnish Research Programme on Nuclear Power Plant Safety 2003–2006

Executive Summary

Edited by
Eija Karita Puska

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VTT, Vuorimiehentie 3, PL 1000, 02044 VTT
puh. vaihde 020 722 111, faksi 020 722 4374

VTT, Bergsmansvägen 3, PB 1000, 02044 VTT
tel. växel 020 722 111, fax 020 722 4374

VTT Technical Research Centre of Finland, Vuorimiehentie 3, P.O. Box 1000, FI-02044 VTT, Finland
phone internat. +358 20 722 111, fax +358 20 722 4374

VTT, Lämpömiehenkuja 3 A, PL 1000, 02044 VTT
puh. vaihde 020 722 111, faksi 020 722 5000

VTT, Värmemansgränden 3 A, PB 1000, 02044 VTT
tel. växel 020 722 111, fax 020 722 5000

VTT Technical Research Centre of Finland
Lämpömiehenkuja 3 A, P.O. Box 1000, FI-02044 VTT, Finland
phone internat. + 358 20 722 111, fax + 358 20 722 5000

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Abstract

Major part of Finnish public research on nuclear power plant safety during the years 2003–2006 has been carried out in the SAFIR programme. The programme has been administrated by the steering group that was nominated by the Ministry of Trade and Industry (KTM). The steering group of SAFIR has consisted of representatives from Radiation and Nuclear Safety Authority (STUK), Ministry of Trade and Industry (KTM), Technical Research Centre of Finland (VTT), Teollisuuden Voima Oy (TVO), Fortum Power and Heat Oy, Fortum Nuclear Services Oy (Fortum), Finnish Funding Agency for Technology and Innovation (Tekes), Helsinki University of Technology (TKK) and Lappeenranta University of Technology (LTY).

The key research areas of SAFIR have been 1) reactor fuel and core, 2) reactor circuit and structural safety, 3) containment and process safety functions, that was divided in 2005 into 3a) thermal hydraulics and 3b) severe accidents, 4) automation, control room and IT, 5) organisations and safety management and 6) risk-informed safety management. The research programme has included annually from 20 up to 24 research projects, whose volume has varied from a few person months to several person years. The total volume of the programme during the four year period 2003–2006 has been 19.7 million euros and 148 person years.

The research in the programme has been carried out primarily by Technical Research Centre of Finland (VTT). Other research units responsible for the projects include Lappeenranta University of Technology, Fortum Nuclear Services Oy, Helsinki University of Technology and RAMSE Consulting Oy. In addition, there have been a few minor subcontractors in some projects.

The programme management structure has consisted of the steering group, a reference group in each of the seven research areas and a number of ad hoc groups in the various research areas.

This report gives a short summary of the results of the SAFIR programme for the period January 2003 – November 2006, and highlights of some major achievements.

Preface

SAFIR, The Finnish Research Programme on Nuclear Power Plant Safety 2003–2006 has continued the tradition of Finnish national research programmes in nuclear area. Organisation of public nuclear energy research in Finland as national research programmes was started in 1989 by the Ministry of Trade and Industry (KTM). Since then national programmes have been carried out in the fields of operational aspects of safety (YKÄ 1990–1994, RETU 1995–1998), structural safety (RATU 1990–1994, RATU2 1995–1998), and in FINNUS 1999–2002 that was the first programme that combined the operational aspects and structural safety. Simultaneously the research was carried out in nuclear waste management programmes (JYT 1989–1993, JYT2 1994–1996, JYT2001 1997–2001, KYT 2002–2005). The research themes of SAFIR will continue in the following research programme SAFIR2010 during the years 2007–2010.

In parallel with the public programmes research has been carried out in the Finnish Fusion Research Programmes (FFUSION and FFUSION2) 1993–2002, programmes on Advanced Light Water Reactor concepts (ALWR) 1998–2003 and a project on component life management 1999–2003, partly funded by the Finnish Funding Agency for Technology and Innovation (Tekes). Currently fusion research continues in the FUSION (2003–2006) and nuclear waste management research in the KYT2010 (2006–2010) programme.

The steering group of SAFIR consists of representatives from Radiation and Nuclear Safety Authority (STUK), Ministry of Trade and Industry (KTM), Technical Research Centre of Finland (VTT), Teollisuuden Voima Oy (TVO), Fortum Power and Heat Oy, Fortum Nuclear Services Oy (Fortum), Finnish Funding Agency for Technology and Innovation (Tekes), Helsinki University of Technology (TKK) and Lappeenranta University of Technology (LTY).

At the beginning of 2004 there was a major change in the funding structure of the programme in comparison with the year 2003 due to a change in the Finnish legislation on nuclear energy. The funding by KTM, STUK, TVO and Fortum was replaced by funding from a separate fund of the State Nuclear Waste Management Fund (VYR). This VYR-funding is collected from the Finnish utilities Fortum and TVO with respect of their MWth shares in Finnish nuclear power plants. The main funding sources of the programme in 2004–2006 have been the State Nuclear Waste Management Fund (VYR) with 2.7 M€ and Technical Research Centre of Finland (VTT) with 1.3–1.5 M€ annually.

The key research areas of SAFIR are 1) reactor fuel and core, 2) reactor circuit and structural safety, 3) containment and process safety functions, that was divided in 2005 into 3a) thermal hydraulics and 3b) severe accidents, 4) automation, control room and IT, 5) organisations and safety management and 6) risk-informed safety management.

SAFIR has been a relatively dynamic research programme allowing inclusion of new projects or extension of the existing projects during the research year. Besides the research done within SAFIR and education of experts via this research, SAFIR has been an important national forum of information exchange for all parties involved.

More information on SAFIR:

http://www.vtt.fi/safir
http://www.vtt.fi/safir2010.

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1. Overview of SAFIR Programme

1.1 The role of SAFIR research programme

SAFIR, The Finnish Research Programme on Nuclear Power Plant Safety 2003–2006 has continued the tradition of Finnish national research programmes in nuclear area.

The programme has been administrated by the steering group that was nominated by the Ministry of Trade and Industry (KTM). The steering group of SAFIR has consisted of representatives from Radiation and Nuclear Safety Authority (STUK), Ministry of Trade and Industry (KTM), Technical Research Centre of Finland (VTT), Teollisuuden Voima Oy (TVO), Fortum Power and Heat Oy, Fortum Nuclear Services Oy (Fortum), Finnish Funding Agency for Technology and Innovation (Tekes), Helsinki University of Technology (TKK) and Lappeenranta University of Technology (LTY). The major partners of SAFIR are shown in Figure 1.

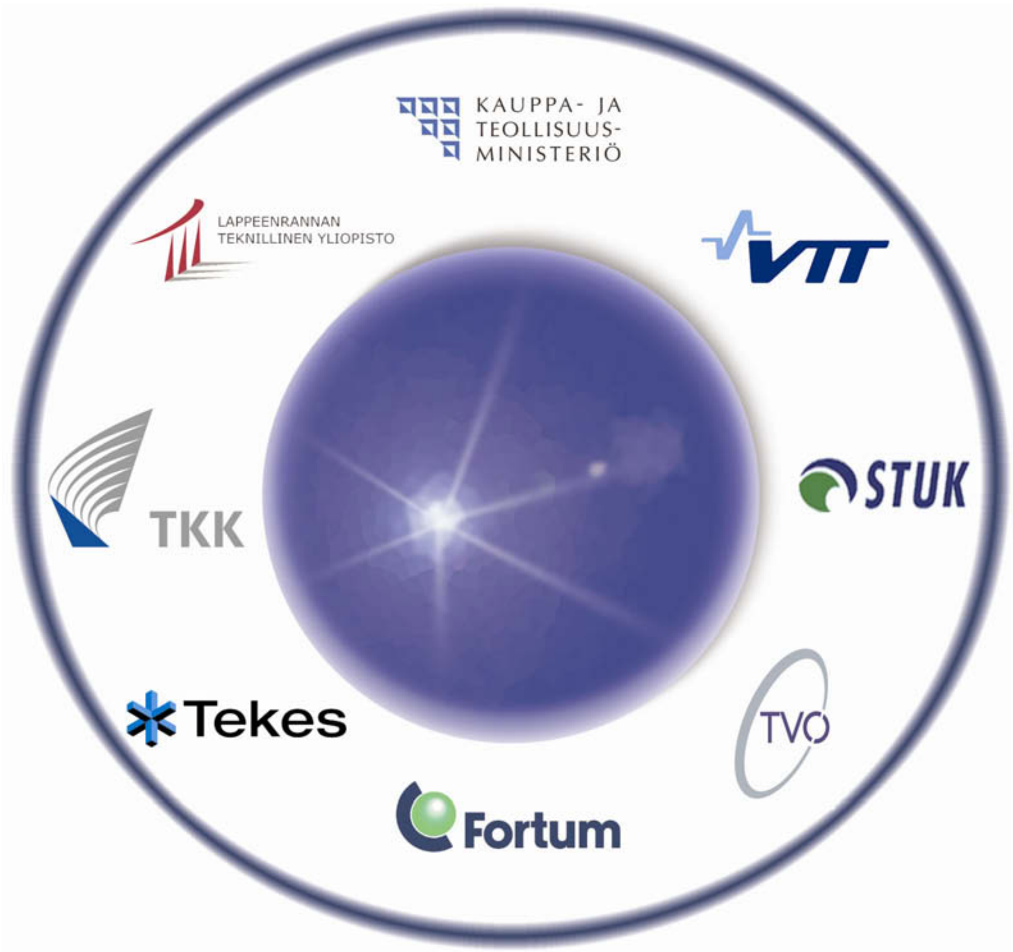

Figure 1. All the key players of the Finnish nuclear field have been represented in the SAFIR steering group.

The role of public nuclear safety research is to provide the necessary conditions for retaining the knowledge that is needed to ensure the continuance of safe and economic use of nuclear power, development of new know-how and participation in international co-operation.

Nuclear safety research in Finland consists of three components: regulatory research, utility research and public research. The roles of these can be illustrated with the 'football field' example in Figure 2. Regulatory research and utility research whose total annual volume exceeds the volume of public research are strictly separated from the public research programme.

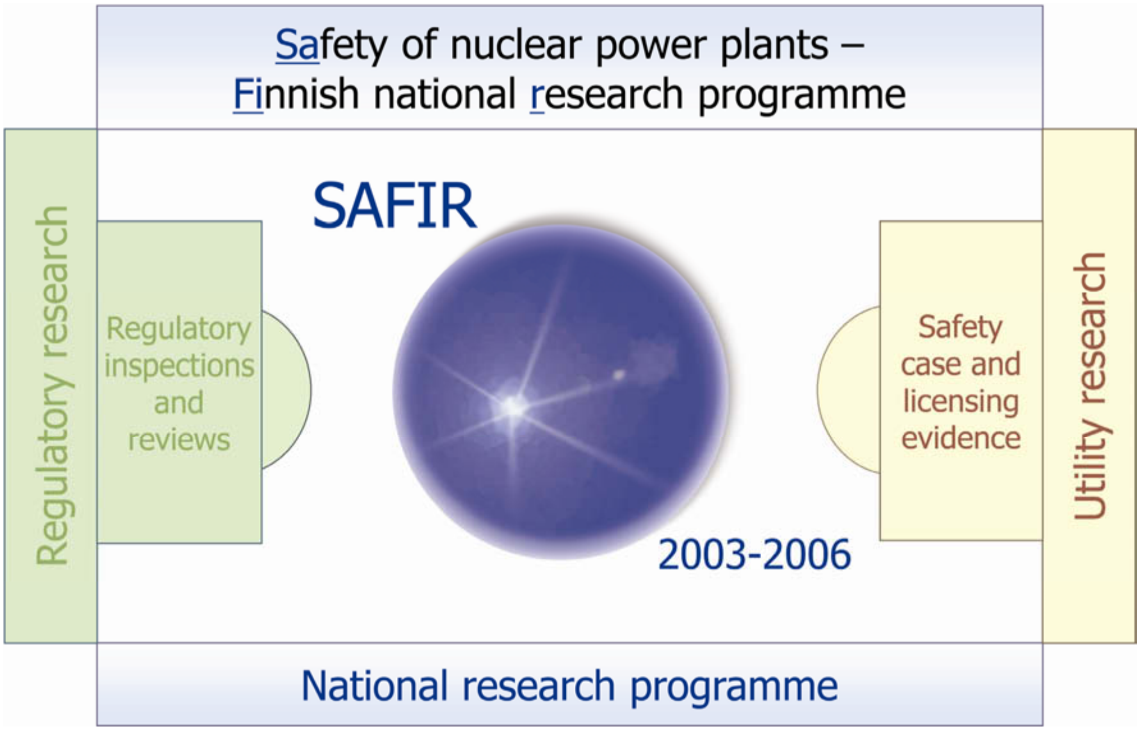

Figure 2. The roles of public, regulatory and utility research in Finland.

Framework plan [1] of SAFIR was made for the period 2003–2006, but it was based on safety challenges identified for a longer time span as well. In the framework plan it was recognized that the safety challenges set by the existing plants and the new plant unit, as well as the ensuing research needs do converge to a great extent. The framework plan defined the important research needs related to the safety challenges, such as the ageing of the existing plants, technical reforms in the various areas of technology and organisational changes. The research that fulfills these needs has been the programme's main techno-scientific task. In addition, the programme had to ensure the maintenance of know-how in the nuclear specific research areas where dynamic research activities are the absolute precondition for safe use of nuclear power.

The SAFIR 2003–2006 programme has taken advantage of the results obtained and lessons learned in the former national research programmes. The programmes in the area of nuclear safety (YKÄ & RATU 1990–1994, RETU&RATU2 1995–1998, FINNUS 1999–2002 and SAFIR 2003–2006) have had the total volume of 75 M€ and 689 person years. According to the final reports of the successive programmes they have produced 2434 publications in various categories and 25 Doctor, 17 Licentiate and 61 Master level academic degrees. The extent of the various programmes is given in Table 1. The Finnish public nuclear safety research will continue in the SAFIR2010 programme during the years 2007–2010.

Table 1. Finnish research programmes in reactor safety area in 1990–2006.

Programme	Volume, M€	Volume, person years	Total number of publications	Academic degrees		
				Dr.	Lic.	M.Sc.
YKÄ 1990–1994	15,4	168	318	6	5	10
RATU 1990–1994	8,2	76	322	1	3	3
RETU 1995–1998	9,8	107	405	3	2	2
RATU2 1995–1998	7,5	60	280	3	4	11
FINNUS 1999–2002	14,4	130	564	6	2	18
SAFIR 2003–2006	19,7	148	545	6	1	17

1.2 Research areas and projects in SAFIR programme

The key research areas of SAFIR have been 1) reactor fuel and core, 2) reactor circuit and structural safety, 3) containment and process safety functions, that was divided in 2005 into 3a) thermal hydraulics and 3b) severe accidents, 4) automation, control room and IT, 5) organisations and safety management and 6) risk-informed safety management.

Reactor fuel and core area covers reactor physics, reactor dynamics and fuel behaviour analysis. The research is done solely with the help of calculational tools, partly with sophisticated tools developed at VTT and partly using tools developed elsewhere. The projects in this area have active contacts with international theoretical and experimental work, such as the OECD Halden Reactor Project, the OECD-IRSN CABRI Project and several other international research projects and working groups.

One of the main goals of the SAFIR programme is the education of the new generation. In this most ‘nuclear-specific’ research area of SAFIR, this task is particularly pronounced. All SAFIR research areas have links both between the various projects in the area and to neighbouring research areas.

During the entire programme there have been two research projects in reactor fuel and core area, the Enhanced methods for reactor analysis (EMERALD) project dealing with reactor physics and dynamics and the High-burnup upgrades in fuel behaviour modelling (KORU) project dealing with the fuel research.

The aim of the **EMERALD** project has been to create a reliable computer code system for reactor calculations in steady-state and transient situations. Such a system has been under development prior to EMERALD already, but new reactor and fuel designs, higher discharge burnups and the need for more accurate results still require the analysis methods to be improved. New neutron transport theory codes based upon the Monte Carlo method (PSG) as well as deterministic solution methods (MultiTrans) have been created. PSG can, for instance, produce group constants for other applications, and in many cases considerably faster than the most commonly used Monte Carlo codes. The nodal BWR core simulator ARES has been further improved and has also been applied to EPR calculations. More accurate cross section libraries and criticality safety studies are developed through international co-operation. The new code ABURN couples the MCNP4C Monte Carlo and the ORIGEN2 isotopic depletion code in order to make burnup credit calculations in criticality safety analysis easier and more accurate. To enable study of some phenomena that could not be simulated by the earlier generation of codes, e.g. coolant flow reversal in a channel, two existing codes, the dynamics code TRAB-3D and thermal hydraulics code SMABRE, have been coupled together using an internal coupling scheme. The thermal hydraulics solver CFDPLIM has been improved to make it more robust, i.e. less sensitive to small disturbances. Sensitivity and uncertainty analysis methods have been studied and utilized in evaluation of uncertainty in the nuclear licensing calculations. Participation in the solution of international benchmark problems has been an essential part of the evaluation of the reliability and accuracy of the reactor analysis codes used in Finland. Finnish participation in the committees and working groups of OECD's Nuclear Energy Agency and other international organizations has also been included in the project. Finally, an important task of EMERALD has been to educate new experts in the field of reactor physics and dynamics, where the need has become quite acute lately.

The work in the **KORU** project has been aimed at creating and maintaining up-to-date methods that provide VTT and the end users of SAFIR results with an independent ability to carry out desired analyses of fuel thermal-mechanical performance. New reactor and fuel types and materials, and evolving operational data – higher discharge burnup goals and other fuel management options – uphold a need for continuous qualification and validation of the single and integral models. Elaborated acceptance guidelines call for availability of sophisticated mechanistic, preferably probabilistic calculation methods. The modelling in the fuel behaviour codes that are in use at VTT has been subject to extensive upgrading in several respects. Major results include:

1) Stream-lining of the active coupling of the advanced thermal hydraulic module GENFLO with successive versions the FRAPTRAN fuel transient code, and vivid application of the tool to the Halden Project IFA-650 LOCA test series and power reactor transients; 2) Complete renewal of the mechanical modelling in FRAPCON, FRAPTRAN, and FRAPTRAN-GENFLO codes using FEM formulation for handling large deformations, frictional fuel-to-clad contact etc; 3) Demonstration of merging an improved mechanistic gas release model as an object-oriented module with the ENIGMA code and its preliminarily validation for future development; 4) Re-correlation of parameters of the fission gas release model in the VTT version of ENIGMA with data from 108 measurements; 5) Basis for a probabilistic procedure to fuel accident behaviour analyses for estimating number of fuel rods that may fail in an accident and 6) Favourable progress in education and training of a new generation of experts.

Reactor circuit and structural safety area covers the studies on the integrity and life time of the entire reactor circuit and the studies of containment building construction, inspection, ageing and repairing. In this area the projects include both experimental and theoretical studies. The projects in this area have also active contacts with international research work, both in EU and elsewhere.

During the programme there have been altogether six research projects in this area. The Integrity and life time of reactor circuits (INTELI) project is a very large one and has extended over the whole time span of the programme. The main objective is to assure the structural integrity of the main components of the reactor circuit of the nuclear power plant and to study the typical ageing mechanisms affecting the integrity of main components during the life-time of the reactor. The main components included in the scope of the project are reactor pressure vessel with nozzles and internals, piping of reactor circuit and other components (steam generators, pumps, valves, pressurizer and heat exchangers). Oxide modelling is studied in the LWR oxide model for improved understanding of activity build-up and corrosion phenomena (LWROXI) project that started in 2004. Containment and concrete structure issues have been studied in a number of projects: Ageing of the Function of the Containment Building (AGCONT, 2003–2004), Participation in the OECD NEA Task Group Concrete Ageing (CONAGE, 2003), Safety Management of Concrete Structures in Nuclear Power Plants (CONSAFE, 2005) and Concrete technological studies related to the construction, inspection and reparation of the nuclear power plant structures (CONTECH) that has continued over the entire programme.

The main objective of the **INTELI** project has been to develop methodology and tools to assure the structural integrity of the main components of the reactor circuit of the nuclear power plant. The main components that have been included in the scope of the project were: 1) Reactor pressure vessel with nozzles and internals, 2) piping of reactor circuit and 3) steam generators.

Concerning the embrittlement mechanisms of reactor pressure vessel and modeling of ageing, the embrittlement mechanisms were identified by analyses of mechanical data and by microstructural analyses.

To ascertain transferability of fracture mechanics test results, low constraint tests were conducted and assessed using finite element analysis. The produced data was added to existing results to develop and verify the Master Curve constraint correction, including both linear-elastic and elastic-plastic fracture mechanics derivations. EAC mechanisms evaluation work in international research programmes was followed and the latest information transferred to Finland through participation to the work of International Co-operative Group on Environmentally Assisted Cracking (ICGEAC). In the area of dosimetry, the adjustment library of the PREVIEW code was improved to include data from two ex-vessel measurements carried out in both Loviisa units. The PREVIEW program was also fine-tuned for production runs on all platforms (at present it works properly only on Unix systems).

A pilot study was carried out to develop and test a semi-quantitative approach for risk-informed in-service inspection (RI-ISI) of piping. From the piping system chosen from TVO primary circuit elements/segments were selected. For these the calculation analyses using probabilistic and risk based analysis procedures developed in the project were used.

Ultrasonic modelling was applied to assess the coverage of critical areas by different transducers in ultrasonic testing and to evaluate the signal that will be measured from a reflector lying in unfavourable orientation at the weld root area. The results are used as a part of technical justification in connection to the inspection qualification.

Two-way coupled Fluid-Structure Interaction (FSI) calculations were carried out by coupling Computational Fluid Dynamics (CFD) and structural analysis codes with ES-FSI and MpCCI codes as linking methods. Two-phase modelling of water and steam was investigated with homogeneous two-phase model and with direct contact condensation model implemented in the Volume Of Fluid (VOF) model. Determination of loads on pressure suppression pool walls from the potential theory of fluid flow was investigated by using the Method Of Images (MOI) and other methods. The calculations were compared to the POOLEX tests conducted at Lappeenranta University of Technology, where air and steam were injected into a water pool.

Fatigue behaviour of stainless steels in PWR applications is of high interest, because extensive life reductions have been measured for some austenitic stainless steels in critical laboratory experiments simulating the oxygen-free PWR water environment. VTT has developed an experimental facility based on advanced bellows loading technology, which has been found to be efficient and reliable in the SAFIR research projects. It can be applied in studying the fulfilment the new requirements set in the YVL guide 3.5 (2002) stating that influence of environment shall be accounted for in fatigue assessment.

Strain controlled constant amplitude and variable amplitude fatigue tests were performed in low oxygen PWR water environment for 316NG and Titanium stabilised stainless steel in 320 °C and 293 °C, respectively. The constant amplitude results were in full compliance with the US and Japanese data, but variable amplitude tests opened new questions to be approached in future research.

Oxidation of different Zirconium-based fuel cladding materials has been studied using a novel thin-layer electrochemical impedance spectroscopic (TLEIS) method in simulated PWR conditions with two different Litium concentrations. The results indicate that the selected experimental approach is capable of detecting differences in oxidation processes in different environments. The results also indicate that the elevated Litium concentration changes the oxidation kinetics on all studied materials. However, the correlation between these changes and the total corrosion rate of the materials is not yet known and further studies are needed and will be performed in the future.

The **LWROXI** project has concentrated on development of modelling tools for activity incorporation into oxide films on construction materials in light water reactor environments. The proposed approach consists of a four-layer model of the oxide formed on stainless steels in nuclear power plant coolants combined with high-temperature adsorption experiments and surface complexation modelling. It features 1) a barrier layer similar to room temperature passive films, which grow via solid state defect transport; 2) an inner layer, that is more defective, its mechanism of growth including both solid state transport and dissolution/redeposition reactions; 3) an outer layer formed by dissolution/precipitation, and 4) a deposit layer the growth of which is dictated by the availability of material able to precipitate onto the outer layer. Support for such a view has been sought in compositional profiles of oxides on stainless steel in nuclear power plant coolant conditions with or without the intentional addition of foreign species, as well as in-situ electrochemical impedance measurements. It is suggested that data on the interaction of oxides with such species could provide an additional way to estimate the kinetic and transport parameters of oxide growth on construction materials in light water reactor primary circuits, as well as to quantify the incorporation of foreign cations from the water into the different film sublayers. The estimated values for the kinetic and transport parameters of the interaction of the oxide with a given coolant species are in general agreement with those available in the literature. These values can be used as input parameters for an integrated activity build-up model that is based on real plant data benchmarking and analysis.

CONTECH project on concrete behaviour has been tightly connected to the research in this area within a large national consortium since 1992. The consortium decides annually the topics to be studied. The funding parties in the consortium include Finnish Road Administration, Finnish Rail Administration and Finnish cities. In 2004 Finnish Nuclear Waste Management Fund (VYR) replaced the Finnish Radiation and Nuclear Safety Authority as a funding party. Also private companies participate in the project occasionally.

CONTECH has consisted of several smaller practical projects dealing with concrete structures. During the years 2003–2006 some 30 projects have been carried out. Examples of topics that have been studied include: internal curing agents of concrete, very rapid hardening grouts, maximum harmless extent and type of rust on reinforcement bars during construction, cathodic protection of reinforcement have been studied, cumulation of salts in concrete, quality requirements and verification methods of concrete protective agents, non-destructive testing methods of reinforcement corrosion, pretreatment of concrete surface for hydrophobic impregnation, quality requirements and verification methods of concrete curing agents, quality requirements of self-compacting concrete, quality requirements of form liners and their usage and effect of form liners and hydrophobic impregnation on chloride permeability. The results obtained are directly applicable in practice: E.g. directions and specifications are made on the basis of the test results.

Thermal hydraulics research area covers simulation of nuclear power plant processess, calculational thermal hydraulics and multiphysics approaches using several codes and thermal hydraulics experiments at Lappeenranta University of Technology (LUT). Multiphysical approaches, strong coupling of experimental and theoretical work and active follow-up and participation in international research programmes are characteristic to the projects in this research area. In this field, with several very ‘nuclear-specific’ projects, fostering of a new generation of experts has a vital role, too.

The projects in this area include The integration of thermal hydraulics (CFD) and finite element (FEM) computer codes in liquid and solid mechanics (MULTIPHYSICS), The Integral code for design basis accident analyses (TIFANY, 2003), APROS modelling of containment pressure suppression systems (TIFANY, 2004), Development of APROS containment model (TIFANY, 2005), Validation of APROS containment model (TIFANY, 2006), Thermal hydraulic analysis of nuclear reactors (THEA), Archiving experiment data (KOETAR) and Condensation pool experiments (POOLEX), that have all continued during the entire programme. In addition there have been two smaller projects, the PACTEL OECD project planning (PACO, 2004) and Participation in Development of European Calculation Environment (ECE, 2005–2006).

The main objective of the **MULTIPHYSICS** project has been to improve modeling capabilities for Fluid-Structure-Interaction (FSI) systems. FSI problems are very common in industry, and there is a growing need to develop simulation tools for these problems. An important application is the structural integrity of nuclear reactor core internals in Design Basis Accident (DBA). The specific case studied in the project has been Large Break Loss of Coolant Accident (LBLOCA) and especially the pressure transient, which is caused by the guillotine pipe break. Tools used for the analysis have been Computational Fluid Dynamics (CFD) and Finite Element stress and strain Analysis (FEA) codes. An original aim of the project was to study the possibility to develop a single numerical model for system geometry with suitable meshing for both CFD and FEA modeling. However, quite soon it was noticed that the common meshing would decrease the quality of both analyses too much. The goal was updated to couple the separate CFD and FEA calculations through internal or external interpolation code. Boundary conditions for the pipe break were calculated using the system codes APROS and TMOC. Fully coupled FSI calculations with FEA code ABAQUS and CFD codes FLUENT and Star-CD were made. Two linking methods were tested: internal ES-FSI model of Star-CD and external interpolation code MpCCI. Both methods worked well and results seemed to be reliable. In addition, the NASTRAN and ABAQUS codes with acoustic elements were tested for the LBLOCA FSI case. Especially ABAQUS results were in good accordance with fully coupled CFD – FEA calculations. Validation against experimental data will be the next step of the MULTIPHYCIS project.

TIFANY project has had annually changing themes related with improvements of the Finnish APROS software in nuclear safety area. In 2005 the main aim of the project was to improve the modeling capabilities of APROS concentrating mainly on the containment model. Several minor improvement targets were selected. In order to confirm the correctness of calculation models, they have to be validated against experimental results. Two experimental sets were selected: an OECD funded International Standard Experiment (ISP) number 35 was calculated to validate the containment internal spray model and two POOLEX suppression pool experiments were calculated to validate water temperature stratification model.

The main goal of the project in 2006 was to validate the APROS containment model against three containment experiments. Two of the experiments were ISP experiments and one was a German spray experiment. ISP-47 experiment is separate effect experiment of steam condensation and the ISP-42 is integral experiment of a severe accident in boiling water reactor containment. As a new feature a pump component was added into the containment model, which allows modeling of the ventilation systems. The Counter current flow limitation (CCFL) in the thermal hydraulic model of APROS was validated against Swedish FRIGG-experiments. This allows application of correlations for other types of nuclear power plants than the Russian VVER.

The **THEA** project has focused on thermal hydraulic analysis of nuclear power plants. The system analysis code APROS was validated by calculating PACTEL NCg, PKL and ROSA experiments. The code capability to simulate conditions where non-condensable gas reduces heat transfer was enhanced. A second order space discretization method was implemented in APROS to better track sharp concentration fronts, like a slug of water with low boron concentration. Young experts have got possibilities to learn modelling of complex thermal hydraulic problems that are found in large test facilities, such as PKL and ROSA.

The 3D porous media code PORFLO was improved so that large problems can be calculated. Capabilities to make containment analysis with CFD code were enhanced. The wall condensation model implemented in the Fluent CFD code was tested by simulating MISTRA experiments. The atmosphere contained air and steam in the first and additionally helium in the second case. The wall condensation and end pressure in both cases was calculated with good accuracy.

At Lappeenranta University of Technology several hundreds of thermal-hydraulic experiments covering a wide range of research topics have been performed with different test facilities since 1975. Data and documents are stored on several media format from CD disks to printed papers. Some of the data and documents are even on media that are not compatible anymore with the hardware and software in use today. In **KOETAR** project, the checked experiment data and documents have been archived in the STRESA database maintained by the Nuclear Safety Research Unit at Lappeenranta University of Technology and also in CD and DVD media. The total amount of the experiments in the database is almost 900. The archived data and documents can be used in nuclear safety research in many ways. For example, several thermal hydraulic and computational fluid dynamics codes can be validated against the checked experiment data in the archive. The database contains also good material for planning and understanding the future experiments and for educational purposes. The data can also be used as a Finnish contribution in international co-operation projects.

The **POOLEX** project has addressed phenomena occurring in large water pools, such as the suppression pool of a BWR, when steam/gas mixture is discharged into liquid. Several experiment series using DN80, DN100 and DN200 blowdown pipes have been carried out with a scaled down test facility designed and constructed at Lappeenranta

University of Technology (LUT). The initial system pressure of the steam source before the blowdown has ranged from 0.2 to 3.0 MPa, the pool water temperature from 8 to 77 ºC and the steam mass flux from 1 to 64 kg/m^{2}s. Wide frequency band instrumentation and data acquisition has been used to record the essential parameters. High-speed video equipment has captured the details of the condensation process at the blowdown pipe outlet. Bubble dynamics, thermal stratification and pressure oscillations due to rapid condensation have been among the issues of interest. Model development of the APROS system code in the TIFANY project and coupling of computational fluid dynamics and structural analysis codes in solving fluid-structure interactions in the INTELI project have been facilitated with the aid of the experiment results from the POOLEX project. A new test facility including an adequate model of the containment upper dry well and withstanding a prototypical pressure has been designed and constructed to further increase the applicability of the experiment results.

Lappeenranta University of Technology (LUT) as well as Technical Research Centre of Finland (VTT) is participating in the Nuclear Reactor Simulations (NURESIM) Integrated Project in the Sixth Framework Programme of EU. The goal of the **ECE** project is to take part in the development and validation process of the new Common European Standard Software Platform for modeling of the problematic two-phase flow simulations of present and next-generation nuclear reactors. The specified participation of LUT and VTT focuses on the thermal hydraulic (TH) part of the project. The participation of VTT has been included in the THEA project. A tailored experiment was carried out for NURESIM in POOLEX project and is used for development of NEPTUNE CFD code in SALOME platform. Testing of the simulation tools started with 2D geometry and is continuing by calculating the POOLEX experiment in 3D geometry. Air as 3rd phase has caused problems in simulation at the moment. A research visit of few months is underway in CEA Grenoble.

Severe accidents research area has included tightly coupled calculational and experimental projects on aircraft crash studies: Wall response to soft impact (WARSI) that was started in 2003 and Impact loaded structures (IMPACT, 2004–2006). Projects considering the more ‘traditional’ issues of severe accidents have included Severe accidents and nuclear containment integrity (SANCY, 2003–2005), Cavity phenomena and hydrogen burns (CAPHORN, 2006), Fission product gas and aerosol particle control (FIKSU, 2003–2004), Behaviour of fission products in air-atmosphere (FIKA, 2005–2006), Development of aerosol models for NPP applications (AMY, 2003–2004) and Emergency preparedness supporting studies (OTUS, 2003).

In the **WARSI** project, in close connection with the IMPACT project, the impact of soft missiles on reinforced concrete target plates has been studied. Impact force time functions of different types of missiles developed during the project have been evaluated by one-dimensional models, the so-called Riera-approach, and by axially symmetric and three dimensional finite element models. In the early stages an affordable soft missile was developed based on thin-walled spirally seamed steel tube. In some cases low-strength concrete was added as payload in the spiral-welded pipe in order to enhance the mass flow effect on the impact force. In further tests water tanks were attached around the spiral-welded pipe missile to simulate the behaviour of fuel during impact. The crushing behaviour of steel missiles can be accurately simulated by a folding mechanism. Strain rate sensitivity of steel has to be taken into consideration. Because steel is a more ductile material than aluminium that is used in aircraft, it was found necessary to perform missile impact tests with aluminium missiles. The break-up behaviour of these missiles changes with impact velocity from folding mechanism to peeling or slicing and for high velocities and especially with water filled missiles to fragmentation into small pieces. Aluminium is less strain-rate sensitive than steel. In experiments determining the impact force the missiles were launched inside the launching tube or above the tube against a rigid force measuring plate. In eight tests the target has been a reinforced concrete plate. The target plate behaviour has been modelled by different non-linear finite element models, which are in this case most convenient because of the target plate support conditions. Good agreement with test results has been found for the deflections and strains recorded from reinforcing bars. Existing data for rigid missiles have been utilized for soft missile impacts by introducing reduction factors in the formulae for scabbing and perforation. Some limited information is available also from these tests but so far only for one span width to plate thickness ratio. From the tests with water tank missiles considerable amount of data about spreading liquid, velocity of spray front, spreading angle, accumulation of fluid, final drop size etc., has been gathered. The first experiences indicate that Fire Dynamics Simulator (FDS) -code is a useful tool to simulate the droplet dispersal. The simulation results also showed that the effect of initial droplet speed, angle of droplet release, and initial drop size are important parameters affecting the formation of the water cloud and final extent of liquid dispersal.

The main goal of the **IMPACT** project was to develop a facility to study the phenomena when an aircraft crashes onto the structure or nuclear power station. To investigate the impact phenomena a new testing facility was built up in a testing hall in VTT. The apparatus is working by pressure accumulator and the missile is accelerated to desired speed by piston, which is moving inside the acceleration tube. The maximum speed of missile used in the tests has been 180 m/s and maximum weight up to 100 kg.

The impact facility has been designed to measure the force-time function of soft missiles during collision. The missiles used in the tests have been hard steel missiles or soft aluminium missiles and some of the tests have been done using missiles filled with water to study also the spreading of the liquid. The force-time function has been measured using rigid force plate in front of the acceleration tube, but impact tests have also performed using concrete wall as a target. Wall tests are done by soft missiles and the main purpose was to study the bending or punching failures of the wall and also missile breaking mechanisms.

SANCY and **CAPHORN** projects have investigated physical phenomena related to severe nuclear accidents with importance to Finnish nuclear power plants. Currently the major topics are the ex-vessel coolability issues, long-term severe accident management and containment leak tightness, hydrogen combustion, and adoption and development of new calculation tools considering also the needs of the future Olkiluoto 3 plant. The projects have employed both experimental and analytical methods.

SANCY project completed the particle bed dry-out heat flux experiments with the STYX facility. The outcome was that coolability of homogeneous bed with prototypical particle size distribution may be more difficult than existing model suggest and the coolability of stratified bed with no heating on top layer in turn may be easier than previous models predict. Increasing bed depth has a limited decreasing effect on dry-out heat flux. Testing of the endurability of various elastomer seal materials against radiation and chemical agitation prevailing in containment after a severe accident revealed that EPDM employed in sealings of electrical cables in Olkiluoto 1 and 2 resists best the aggressive environment following a severe accident. New analytical tools have been acquired and implemented at VTT. These tools include NUCLEA/GEMINI code applied to estimation of fluidity of various in-vessel or ex-vessel core melt mixtures for all Olkiluoto and Loviisa units. Further in the CAPHORN project the TONUS hydrogen mixing and combustion code developed at CEA/France has been acquired for applications of Olkiluoto 3 plant and a 2D-code WABE developed at IKE Stuttgart for advanced particle bed coolability analyses in plan geometry. One important task of CAPHORN project was also the design of molten metal-concrete experiments and construction of the HECLA facility. The scoping test was run in the frame of CAPHORN, the more challenging part of the test matrix being a task of the future COMESTA project.

FIKSU and FIKA projects have been a combination of studies related to severe accident issues. In the main part of the work the release, transport and speciation of ruthenium in conditions simulating an air ingress accident was experimentally studied. The objective of the experiments was to quantify the fraction of ruthenium released in gaseous form into the containment in various conditions. Experiments conducted in international PHEBUS FP program were also followed up. The PHEBUS FP experiments simulate the major aspects of a severe accident. Data from the tests have been applied to refine models of core degradation and fuel relocation, hydrogen production, and fission product specification. Investigations carried out in the frame of

SAFIR program have provided important information on mass balance and primary circuit transport of the fission products as well as means to improve aerosol and iodine measurements in the experiments.

Phebus FP experiments also pointed out areas, where current understanding is not sufficient. Such areas like fission product chemistry in the primary circuit or aqueous and gaseous chemistry of iodine within the reactor containment are currently addressed in the frame of International Source Term Program.

In ARTIST experimental program conducted at Paul Scherrer Institute (2003–2007), Switzerland, the retention of aerosol-borne fission products in the steam generator (SG) secondary is studied. In SAFIR program VTT provides aerosol measurement instrumentation and expertise for integral ARTIST-experiments.

The research in SAFIR in the areas **automation, control room and IT, organisations and safety management and risk informed safety management** concentrates on the nuclear-specific problems. A typical feature of all these research areas is that the majority of total research activities both in Finland and abroad are directed to non-nuclear applications and that same tools and methods can be used quite extensively both in nuclear and in non-nuclear research problems.

Automation, control room and IT research area has focused on the new technologies that are emerging at nuclear power plants both via new plants and via renewal of automation and control rooms in the existing plants.

The research projects in this area have contained the Interaction approach to development of control rooms (IDEC) project that has continued over the whole programme and a number of smaller projects with shorter duration including: Application possibilities of systematic requirements management in the improvement of nuclear safety in Finland (APSREM, 2003), Influence of RoHS-directive to reliability of electronics – preproject (ROVEL, 2004), Software qualification – error types and error management in software life-cycles (QETES, 2005–2006) and Influence of Whiskers to Reliability of Electronics, Prestudy (WHISKE, 2005).

The **IDEC** project aimed at formulating a scientifically founded method for the evaluation of human-system interfaces of complex industrial systems. This study anticipated the needs for knowledge, methods and know-how concerning human factors engineering (HFE) evaluation that the modernisation of the Finnish nuclear power plants and their control rooms would bring. Evaluation of both interface and the design process has been considered. The resulting evaluation framework is called Contextual Assessment of Systems Usability (CASU).

In the development process of CASU three different methodical approaches were used: review of guidelines and standards, case studies on design processes and experimental testing. The third part the project was accomplished in collaboration with the OECD Halden Reactor project, and also with University of Toronto and Electricité de France. The main result of the study is the formulation and first practical testing of the framework, the Contextual Assessment of Systems Usability (CASU). IDEC project has created the structure, the indicators, empirical analysis methods of practices, and the criteria of evaluation. The novel feature of the method is to extend the evaluation basis. For this a new concept the "systems usability" has been developed. It emphasises the evaluation of the interface in the context of fulfilling the targets and safety critical functions of the whole NPP production. The demands of the production become overt in the operators' core task, including demands based on objectives, and intrinsic constraints and possibilities of work and specific context-dependent tasks. The structure of the method consists of three phases: modelling, data collection, data analysis, and assessment of systems usability. Systems usability is evaluated against the tool's ability to promote core-task oriented practices, and against its completeness in fulfilling three central functions of a tool, i.e. instrumental, psychological and communicative. Hence the control room and its interfaces should be effective and efficient for process control, be fit for human use and promote seamless human-technology and team collaboration, and create shared awareness and meaningful mastery of work. The future work will be oriented to validate the CASU method in further steps of control room evaluation.

The Software qualification – error types and error management in software life-cycles project **QETES** has given a new approach to classify software defects. The approach is based on the three aspects of computer semiotics: syntactic, semantic and pragmatic. Previously several defect classifications have been proposed in software engineering, but most of them do not focus on the defect detection and prevention. The new approach supports to identify, which type of defects can be eliminated by each of the available means. This makes the qualification of safety critical software more effective and helps to assess the coverage of fault elimination. The proposed computer semiotic classification of defects has been validated by studying a number of incidents involving software errors.

As syntax defects are reliably caught by compilers, some semantic and pragmatic defects are more cumbersome. Those semantic defects that cause run time errors can be reliably caught by on-line self checks, but for instance domain factual errors are independent of software development and implementation; they can be caught only by expert assessments and diverse redundancy. Pragmatic errors cover several kinds of misinterpretations between components as several previous incidents due to software errors have been indicated. Even if pragmatic errors can be detected by experimental means and self diagnostics, the best means to avoid them is careful study of interaction behavior of components that have different features.

Organisations and safety management research area focuses on the organisational culture and management of change and on the tacit knowledge involved. The expertise of this research area is used also in the neighbouring areas in questions related to control rooms and automation and in research related to fires at NPPs.

The research has been carried out in two projects: Organisational culture and management of change (CULMA) that has continued over the whole programme and Disseminating tacit knowledge in organisations (TIMANTTI, 2004–2006). Similarly to other research areas, the projects involved also participation in international research and working groups.

The goal of the **CULMA** project has been to clarify the effects of organizational factors on nuclear safety. The project has focused on three themes; organizational changes and new ways of organizing work in the nuclear industry, theory of high reliability organizations, and methodology for assessing organizational culture (CAOC). Case studies were carried out in both Finnish NPPs' maintenance organizations and TVO nuclear power plant engineering. Furthermore, collaboration with Swedish researchers and power companies was carried out in connection to all the themes. In the project, various work psychological theories and models were tested and developed with the aim of tailoring appropriate methods for research and development in safety critical environments. During the project a model of the elements (structure, norms, personnel's conceptions) affecting the functioning of the organization was created. The model offers a basis for the development of various human performance improvement tools and assessment of the safety impacts of organizational factors such as structural changes or organizational climate. The models and specific measures that have been developed have been published in international conferences and scientific publications, and the organizational culture assessment methodology (CAOC) has been submitted for review as an academic dissertation. A Finnish language book has been written on the special characteristics of safety critical organizations. In the book a conceptualization is made of the organizational and psychological challenges that these organizations must solve. The book is available also in English and can be used as training material in human factors issues.

The **TIMANTTI** project has focused on two main topics: 1) defining tacit knowledge in nuclear power plants and 2) developing practices for preserving and sharing tacit knowledge. These topics have been studied in three contexts: mechanical maintenance, technical design and control room/operator training. The project has contributed to identifying the critical tacit knowledge as well as the prerequisites for preserving it. As the outcome the DIAMOND (TIMANTTI) model of tacit knowledge was created. In addition, several methods for tacit knowledge sharing found in literature were analyzed in terms of their applicability in nuclear power plants. This analysis together with good practices identified in studied context produced a process model of apprenticeship. The results of the project are applicable also more widely in the nuclear power sector to recognize critical tacit knowledge and support the preserving of tacit knowledge in nuclear power context.

Risk-informed safety management means use of information from probabilistic safety assessment (PSA) to support decision making in various contexts. The expertise on risk-informed safety assessment methods are used also in some projects in other research areas in SAFIR.

The research area has included two projects that have continued over the entire programme: Potential of Fire Spread (POTFIS) and The Principles and Practices of Risk-Informed Safety Management (PPRISMA). The third project in the area is Assessment of smart device software (ASDES, 2005–2006) that has strong links to the automation research area.

In the **POTFIS** project the central goal for fire research has been continuing the avenue opened during the former national research programme FINNUS to develop deterministic and stochastic submodels to the same level as other branches of PSA. The major strategic problem during SAFIR was the ability to predict potential of fire spread in given scenarios. Three subprojects were started: (1) flame spread experiments on cables, (2) fire spread modelling, and (3) reliability of active and operative fire protection. In the later part of the project, participation in the international OECD project PRISME was added. In subproject (1) experiments at various scales on vertical flame spread on and autoignition of thin solids were carried out. New flame spread measuring instruments were proposed: a new 2 m vertical sample test rig for flame spread velocity as a function of initial temperature, now under construction, and double cone calorimeters for energy and mass balance, preliminarily tested during the project. In subproject (2), a new vertical flame spread model was depicted using literature studied, by small scale experiments from subproject (1) and by direct numerical simulation (DNS) solving of Navier-Stokes equation. A new Two-Model Monte Carlo (TMMC) technique was developed for the probabilistic analysis using computationally intensive simulation models. TMMC was applied on the analysis of fires in the NPP electronics and cable rooms. Subproject (2) included also participation in International Collaborative Project to Evaluate Fire Models for Nuclear Power Plant Applications (ICP). In subproject (3) modelling efforts of active fire protection devices that were started during the previous research programme FINNUS were completed.

The **PPRISMA** project has dealt with both risk-informed decision making and methods for risk assessment. In addition, the research activities are carried out through international co-operation.

The status and experience with the technical basis and use of probabilistic risk criteria of nuclear power plants in Finland and Sweden have been compiled. A process model for planning a risk informed and cost-effective maintenance programme has been constructed. A systematic method has been developed for the analysis of human failures from maintenance history information with emphasis on identification of possible common cause failures. An expert collaboration method developed provides a scenario analysis and a systemic network analysis for considering the controllability of fire situations from the situation assessment and co-operation point of view. In risk-informed categorization, the systems, structures and components of NPP are classified based on PSA information, providing a complementary view to deterministic safety classification.

A quantitative reliability estimation method of computer-based systems operating in safety-critical applications was developed. The method is based on utilization of expert opinions and Bayesian inference and has been tested in two cases on software reliability of a motor protection relay. In another task, a Markov model based modelling tool was developed for reliability analysis of simple systems with long term mission time, e.g., residual heat removal function which can be needed months in an accident case.

The overall objective of the **ASDES** project has been to develop an approach to the assessment of such smart device software that takes into consideration: 1) the particular issues of assessing COTS (commercial off-the-shelf) and the design and accessibility of smart devices, 2) regulatory context of the nuclear industry in Finland (e.g. YVL guides 2.0, 2.1, 2.7, 2.8, 5.5) and 3) current practices of software assurance developed in Finland and more widely in the UK and European projects.

The approach developed is goal based method that defines claims, elaborates and apportions them to smart devices and components and then creatively identifies the arguments required to show these claims. Then, one has to assess whether the claims are satisfied in the light of available evidence. The goal based view emphasises the claim and argument structure. This is important as, according to the developers, the experience is that these aspects that are not well articulated.

During the process of developing the approach, the relevance of the YVL Guide clauses was assessed. Based on the assessment, a spreadsheet providing a commentary on each clause was produced. The spreadsheet can be used to assess the completeness of safety case and by integrating into the case, it is possible to provide a “compliance case” with respect to YVL Guidelines.

1.3 Statistical information

The research programme has included annually from 20 up to 24 research projects, whose volume has varied from a few person months to several person years. The total volume of the programme during the four year period 2003–2006 is 19.7 million euros and 148 person years. The research projects and administration (SAHA) have been collected in Table 2. The research has been carried out primarily by the Technical Research Centre of Finland (VTT). Other research units responsible for the projects include Lappeenranta University of Technology, Fortum Nuclear Services Oy, Helsinki University of Technology and RAMSE Consulting Oy. In addition, there have been a few minor subcontractors in some projects.

Table 2. The research projects of SAFIR in 2003–2006.

			Funding (thousand Euro)				Volume (person-years)				
Group	**Project and principal research organisation**	**Acronym**	**2003**	**2004**	**2005**	**2006**	**2003**	**2004**	**2005**	**2006**	**Total**
1.						plan	y	y	y	y	
	Enhanced methods for reactor analysis *VTT*	EMERALD	487	572,9	568,1	583	4,36	4,96	4,4	4,45	18,17
	High-burnup upgrades in fuel behaviour modelling *VTT*	KORU	220	295,7	285,5	293	2,11	3,04	2,9	2,43	10,48
2.											
	Integrity and life time of reactor circuits *VTT*	INTELI	1019	1082	1215,3	1149	7	7,60	7,4	6,99	28,99
	LWR oxide model for improved understanding of activity build-up and corrosion phenomena *VTT*	LWROXI	-	86	64,1	92	-	0,62	0,3	0,78	1,7
	Ageing of the function of the containment building *VTT*	AGCONT	13	34	-	-	0,1	-	-	-	0,1
	Participation in the OECD NEA task group concrete ageing *VTT*	CONAGE	9,5	-	-	-	0,07	-	-	-	0,07
	Safety Management of Concrete Structures in Nuclear Power Plants VTT	CONSAFE	-	-	17,5	-	-	-	0,1	-	0,1
	Concrete technological studies related to the construction, inspection and reparation of the nuclear power plant structures *VTT*	CONTECH	100,5	107,5	144,6	9,2	0,7	0,85	1,3	0,88	3,73
3.											
a	The integration of thermal hydraulics (CFD) and finite element (FEM) computer codes in liquid and solid mechanics *Fortum Nuclear Services*	MULTI-PHYSICS	-	117	154	98	-	0,69	1,2	1,05	2,94
a	The integral code for design basis accident analyses *Fortum Nuclear Services*	TIFANY	203,3	209,6	194,3	116,8	1,53	1,73	1,4	1,00	5,66

a	Thermal hydraulic analysis of nuclear reactors *VTT*	THEA	165,8	165	206,3	259	1	0,95	1,4	2,00	5,35
a	Archiving experiment data *Lappeenranta University of Technology*	KOETAR	60	60,6	50,3	40	0,6	0,60	0,5	0,38	2,08
a	Condensation pool experiments *Lappeenranta University of Technology*	POOLEX	163,9	258,5	271,3	292	2	2,90	3,4	2,10	10,4
	PACTEL OECD project planning *Lappeenranta University of Technology*	PACO	-	80,3	-	-	-	0,19	-	-	0,19
a	Participation in development of European calculation environment *Lappeenranta University of Technology*	ECE	-	-	53,2	50	-	-	0,7	0,67	1,37
b	Wall response to soft impact *VTT*	WARSI	138,4	140,8	152,6	170,5	1,2	1,24	0,7	1,00	4,14
b	Impact tests *VTT*	IMPACT	-	221,5	153,0	220	-	1,50	1,1	1,52	4,12
b	Severe accidents and nuclear containment integrity *VTT*	SANCY	314	306	305,6	-	1,56	1,70	2,8	-	6,06
b	Cavity phenomena and hydrogen burns *VTT*	CAPHORN	-	-	-	320,25	-	-	-	2,00	2
	Fission product gas and aerosol particle control *VTT*	FIKSU	112	56,1	-	-	1	0,50	-	-	1,5
b	Behaviour of fission products in air-atmosphere VTT	FIKA	-	-	250,6	315	-	-	2,4	2,62	5,02
	Development of aerosol models for nuclear applications *Fortum Nuclear Services*	AMY	150	148,4	-	-	1,9	1,43	-	-	3,33
	Emergency preparedness supporting studies *VTT*	OTUS	50	-	-	-	0,41	-	-	-	0,41
4.											
	Interaction approach to development of control rooms *VTT*	IDEC	140	196	197	17,5	1,1	1,34	1,4	1,67	5,51
	Application possibilities of systematic requirements management in the improvement of nuclear safety in Finland *RAMSE Consulting*	APSREM	50	-	-	-	0,44	-	-	-	0,44

	Influence of RoHS-directive to reliability of electronics – preproject *VTT*	ROVEL	-	20	-	-	-	0,11	-	-	0,11
	Software qualification – error types and error management in software life-cycle VTT	QETES	-	-	74,1	74,94	-	-	0,5	0,43	0,93
	Influence of Whiskers to Reliability of Electronics, Prestudy VTT	WHISKE	-	-	24,5	-	-	-	0,1	-	0,1
5.											
	Organisational culture and management of change *VTT*	CULMA	201	210	178,9	184	1,45	1,52	1,6	1,43	6,00
	Disseminating tacit knowledge in organisations (preproject 04) *Helsinki University of Technology*	TIMANTTI	-	27,3	69,8	52	-	0,43	1,1	0,67	2,20
6.											
	Potential of fire spread *VTT*	POTFIS	158,5	158	158,0	208	0,7	1,00	1,0	1,24	3,94
	Principles and practices of risk-informed safety management *VTT*	PPRISMA	245	248,5	215	224,9	2,07	2,05	1,6	1,50	7,22
	Assessment smart device software VTT	ASDES	-	-	80	80	-	-	0,2	0,57	0,77
0.	SAFIR Administration and information (2002–2003) *VTT*	SAHA	107,9	125,1	112,6	194,73	0,81	0,75	0,5	0,95	3,01
	Total		**4108,76**	**5058,0**	**5197,6**	**5360,6**	**32,11**	**37,96**	**40,0**	**38,31**	**148,38**

Distribution of total funding in the SAFIR research areas in 2003–2006 is shown in Figure 3. At the beginning of 2004 there was a major change in the funding structure of the programme in comparison with the year 2003 due to a change in the Finnish legislation on nuclear energy. The funding by KTM, STUK, TVO and Fortum was replaced by funding from a separate fund of the State Nuclear Waste Management Fund (VYR). This VYR-funding is collected from the Finnish utilities Fortum and TVO with respect of their MWth shares in Finnish nuclear power plants. The main funding sources of the programme in 2004–2006 have been the State Nuclear Waste Management Fund (VYR) with 2.7 M€ and Technical Research Centre of Finland (VTT) with 1.3–1.5 M€ annually. Two projects have obtained funding from Tekes instead of the VYR-funding. The distribution of the VYR-funding to the various research areas is shown in Figure 4. Distribution of funding and person years in the seven research areas of SAFIR in 2006 have been illustrated in Figures 5 and 6, respectively.

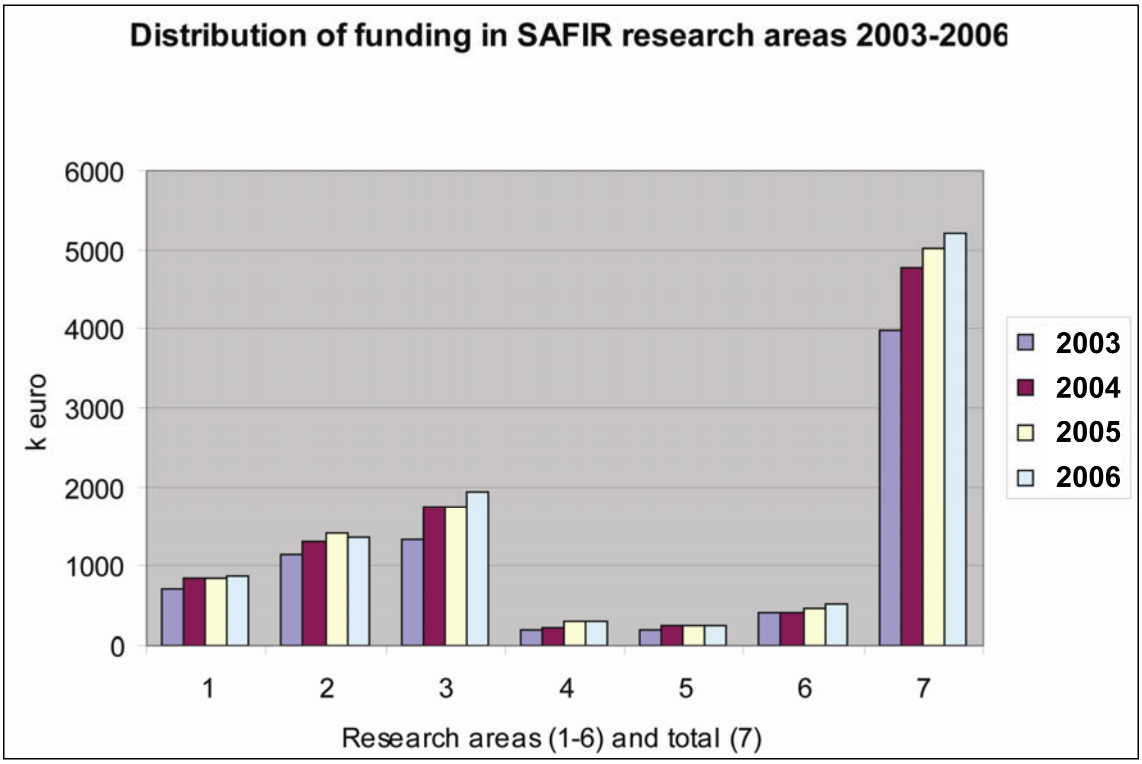

Figure 3. Distribution of total funding in the SAFIR research areas in 2003–2006.

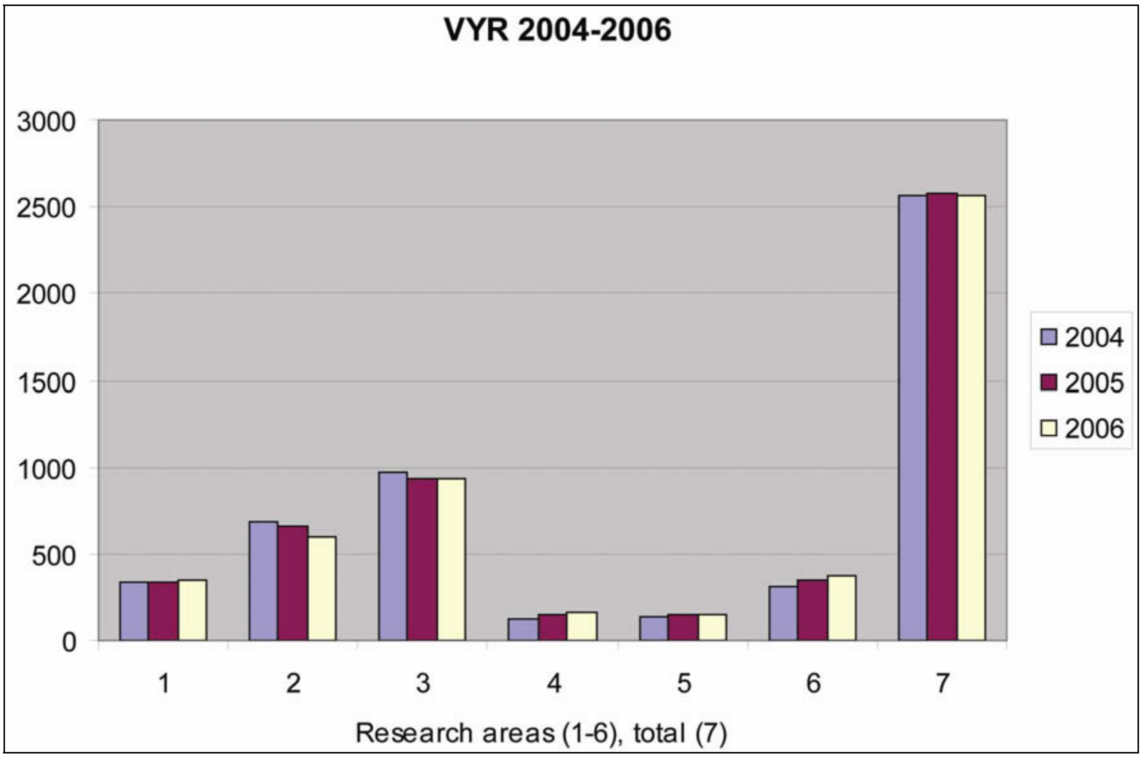

Figure 4. Distribution of VYR funding in the SAFIR research areas in 2004–2006.

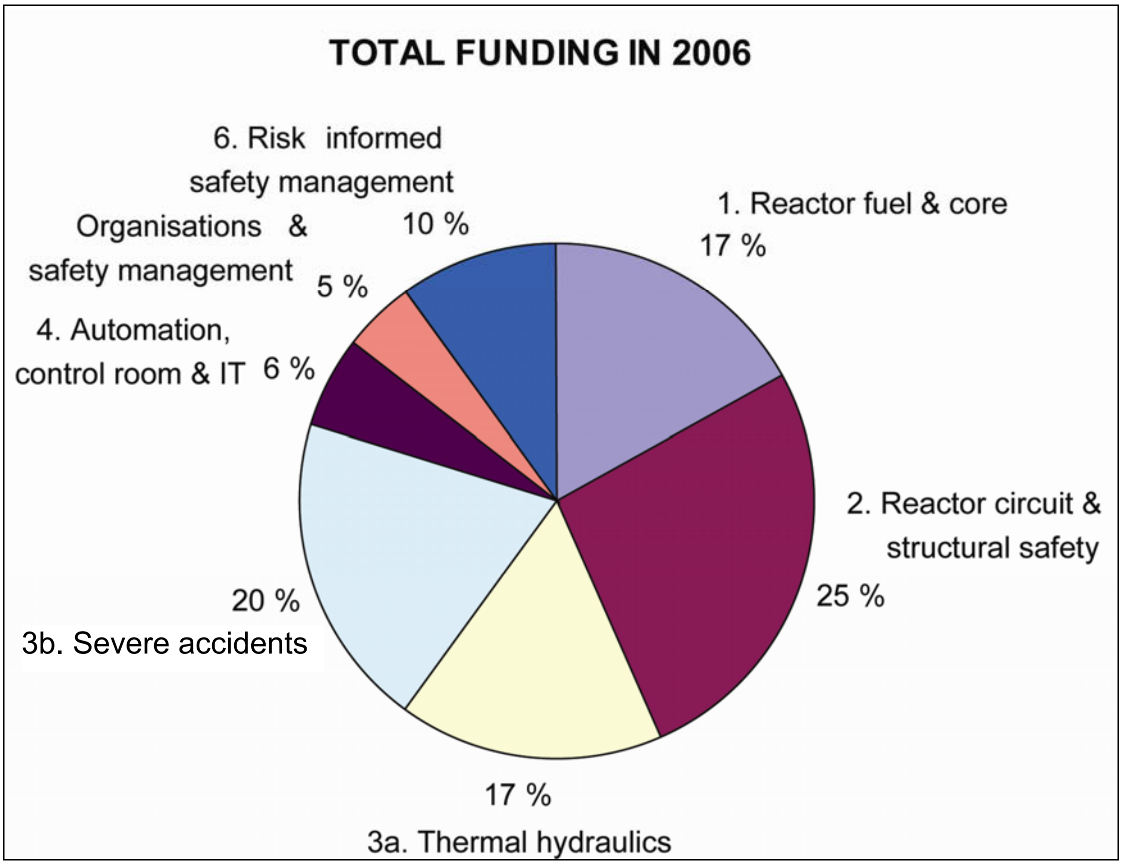

Figure 5. Distribution of funding in the SAFIR research areas in 2006.

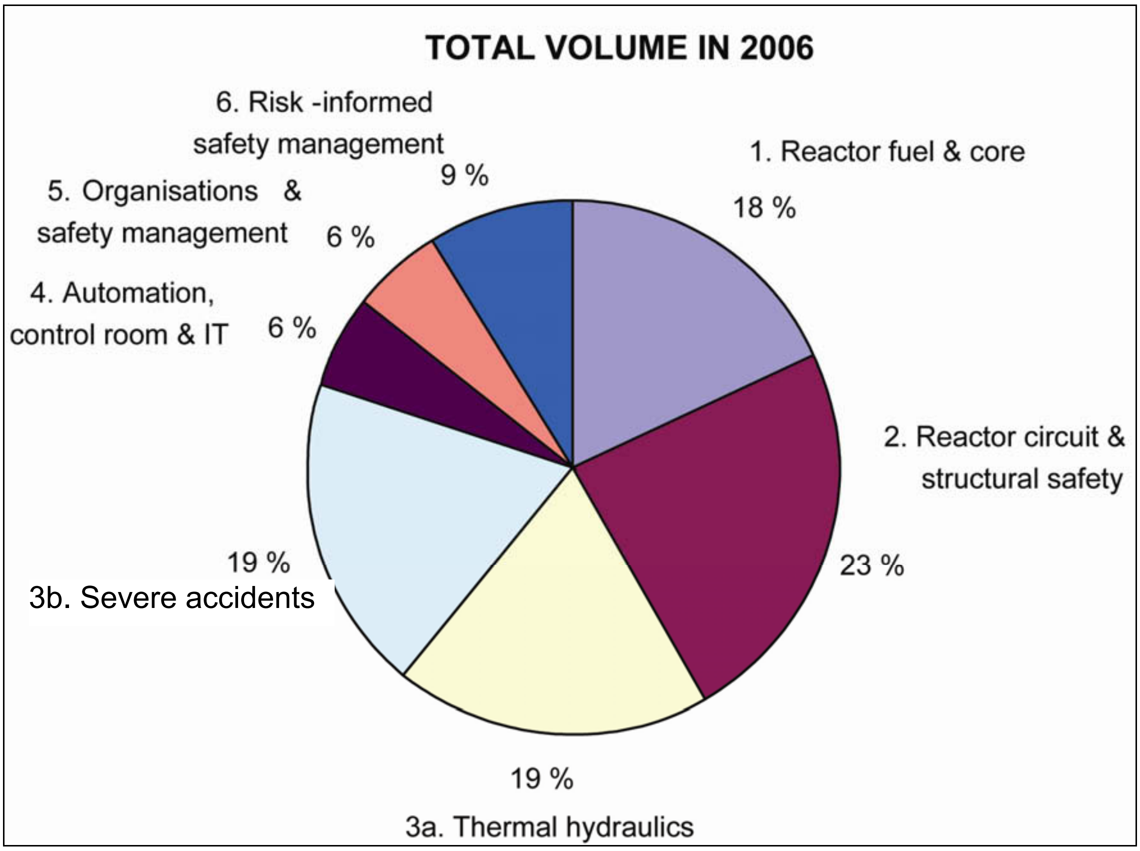

Figure 6. Distribution of person years in the SAFIR research areas in 2006.

The programme has produced 545 publications in 2003–2006. Major part of the publications consisted of conference papers and extensive research institute reports. The number of scientific publications as well as the total number of publications varied greatly between the projects, as indicated in Table 3. The average number of publications is 3,7 per person year, and the average number of scientific publications is 0,3 per person year. Some projects have deliberately aimed at publication of the results as extensive research institute reports that are found to be more useful to the end-users than scientific publications which has to be taken into account when judging the numbers of publications in different categories.

Table 3. Publications in the SAFIR projects in 2003–2006.

Project	**Scientific**	**Conference papers**	**Res. inst. reports**	**Others**	**Total**	**Volume pers. year**
EMERALD	7	37	21	3	68	18,17
KORU	0	4	9	3	16	10,48
INTELI	15	37	34	0	86	28,99
LWROXI	1	3	2	0	6	1,70
AGCONT	0	0	3	0	3	0,1
CONAGE	*0*	*0*	*0*	*0*	*0*	*0,07*
CONSAFE	0	0	0	0	0	0,10
CONTECH	0	1	17	5	23	3,73
MULTIP	0	3	2	3	8	2,94
TIFANY	0	1	25	0	26	5,66
THEA	1	6	6	3	16	5,35
KOETAR	0	1	3	1	5	2,08
POOLEX	0	2	8	6	16	10,40
PACO	0	0	2	0	2	0,19
ECE	0	0	0	4	4	1,37
WARSI/IMPACT	0	4	8	1	13	8,26
SANCY/CAPHORN	1	4	17	1	23	8,06
FIKSU/FIKA	3	15	10	1	29	6,52
AMY	0	2	6	3	11	3,33
OTUS	*0*	*0*	*3*	*0*	*3*	*0,41*
IDEC	5	14	7	1	27	5,51
APSREM	*0*	*0*	*2*	*0*	*2*	*0,44*
ROVEL	0	0	2	0	2	0,11
QETES	0	0	2	1	3	0,93
WHISKE	0	2	1	2	5	0,1
CULMA	6	11	7	9	33	6,00
TIMANTTI	4	7	0	2	13	2,20
POTFIS	7	20	6	14	47	3,94
PPRISMA	1	14	23	7	45	7,22
ASDES	0	0	1	0	1	0,77
SAHA	0	0	8	1	9	3,01
TOTAL	**51**	**188**	**235**	**71**	**545**	**148,38**

The programme has produced so far 6 Doctoral degrees, 1 Licentiate degree and 17 Master's degrees, as indicated in Table 4.

Table 4. Academic degrees awarded in the projects.

Project	Doctor (DTech, PhD)	Licentiate (LicTech, LicPhil)	Master (MScTech, MSc, MA)	Total
EMERALD	1	1	1	3
KORU	-	-	1	1
INTELI	2	-	1	3
THEA	-	-	3	3
POOLEX	-	-	3	3
ECE	-	-	1	1
WARSI	-	-	1	1
SANCY	-	-	1	1
AMY	-	-	2	2
FIKA	1	-	-	1
IDEC	-	-	1	1
POTFIS	1	-	-	1
PPRISMA	1	-	2	3
Total	6	1	17	24

1.4 Administration, seminars and international evaluation

The programme management bodies, the steering group and the six, from 2005 onwards seven reference groups, have met on regular basis 3–4 times annually. The ad hoc groups that have a vital role for some projects have carried out successfully their tasks. The ad hoc groups have met upon the needs of the specific project. The most active and also most essential ad hoc group for the successful outcome of the project has been the WARSI-IMPACT ad hoc group. Figure 7 illustrates the structure of the SAFIR programme with the research projects forming the hot red core of the programme, the seven reference groups and the various ad hoc groups having the principal responsibility of scientific guidance and surveillance of the various research projects, as depicted with the yellow layer encircling the red core. The steering group, depicted as the blue layer, has administrated the entire research programme thus keeping the SAFIR 'jewel' together. The programme is managed by the coordination unit VTT, the programme director, the project co-ordinator and the project managers of the individual research projects.

Figure 7. The three-layer structure of SAFIR programme with projects (red), reference and ad hoc groups (yellow) and steering group (blue).

The information on the research performed in SAFIR has been communicated formally via the quarterly progress reports, the annual plans and annual reports [2–7] of the programme and the www-pages of the programme. Additional information has been given in seminars organised by various research projects. The detailed scientific results have been published as articles in scientific journals, conference papers, and separate reports. Brochure on the SAFIR programme was published in 2004. Interim Seminar of the programme was arranged in January 2005 with 118 participants and 22 scientific presentations and panel discussions, and Interim Report [8] was produced.

An independent international evaluation of the SAFIR-programme was ordered by KTM. The panel of three members carried out its evaluation by reviewing copies of relevant documents and, during a one-week period 19–24 March 2006, meeting with key individuals. The results of the panel were provided as general conclusions, responses to questions posed by KTM, challenges and recommendations and comments on specific projects in each subject area [9].

As a part of the planning process of the next research programme, a strategy seminar, SAFIR2010, was organised in April 2006 in Innopoli 2. Approximately 90 persons took part in the seminar. The new programme, SAFIR2010, starts at the beginning of January 2007 [10].

In addition to conducting the actual research according to the yearly plans, SAFIR has been an efficient way of information exchange with all organisations operating in the nuclear energy sector and as an open discussion forum for participation in international projects, allocation of resources and in planning of new projects.

Appendix A contains selected highlights of the SAFIR projects. A more detailed description of the research programme and its accomplishments can be found in the SAFIR Final Report [11].

1.5 Acknowledgements

The results of the SAFIR programme have been produced by all those involved in the actual research projects. Their work is highly esteemed.

The contributions of project managers and project staff that form the essential contents of this report are acknowledged with gratitude.

The work of the persons in the Steering Group, Reference Groups and Ad Hoc Groups that has been carried out with the expense of their home organisations is highly appreciated.

Eija Karita Puska

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3 Puska, E. K. SAFIR, The Finnish Research Programme on Nuclear Power Plant Safety, Annual Report 2003. Espoo: VTT Processes Project Report PRO1/P7001/04. 58 p. + app. 119 p.

4 Puska, E. K. SAFIR, The Finnish Research Programme on Nuclear Power Plant Safety, Annual Plan 2004. Espoo: VTT Processes Project Report PRO1/P7002/04. 22 p. + app. 130 p. Puska, E. K., SAFIR, The Finnish Research Programme on Nuclear Power Plant Safety, Annual Report 2004. Espoo: VTT Processes Project Report PRO1/P1014/05. 69 p. + app. 138 p.

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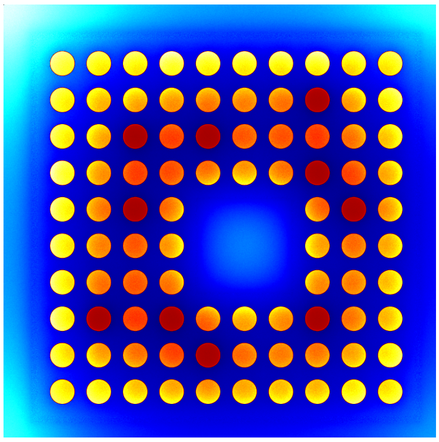
Appendix A: Highlights of SAFIR

Reactor physics analysis
PSG (Probabilistic Scattering Game)

- New neutron transport code based upon the Monte Carlo method
- Generates group constants and other parameters for diffusion and nodal codes
- Multiplication factors and power distributions in reactor fuel and core calculations
- In many applications 2–16 times faster than commonly used Monte Carlo codes
- Accuracy typically better than 0.5 % formost neutron cross sections, 100 pcm for multiplication factor

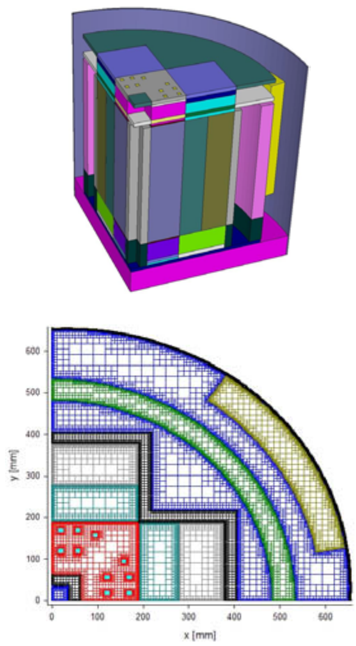


Power distribution in BWR bundle.

Contact person: Jaakko Leppänen, VTT, jaakko.leppanen@vtt.fi

Reactor physics analysis
MultiTrans

- New deterministic 3D radiation transport code based on Simplified P_3 approximation of the transport equation combined with advanced tree multigrid technique
- Adaptive meshing is utilised in generating grid directly from CAD model
- Multigrid technique is used to accelerate the iteration, which enables fast solution of multiplication eigenvalue, multigroup neutron spectra and dosimetric reaction rates
- SP_3 approximation has only limited accuracy, especially in deep penetration problems or in problems with low-density regions, but is superior to simple diffusion theory

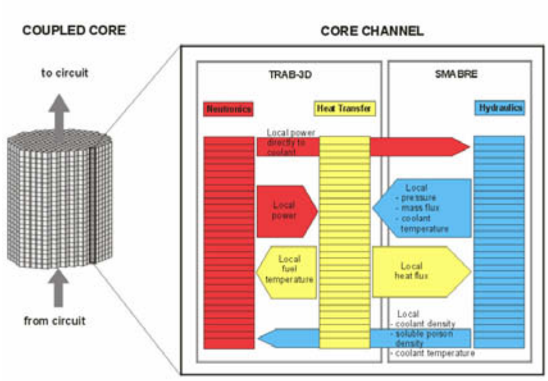


Modelling of the VENUS-2 reactor for benchmark calculations.

Contact person: Petri Kotiluoto, VTT, petri.kotiluoto@vtt.fi

PWR and BWR transient and accident analyses
The TRAB-3D reactor dynamics code and SMABRE thermal hydraulics system code coupled internally

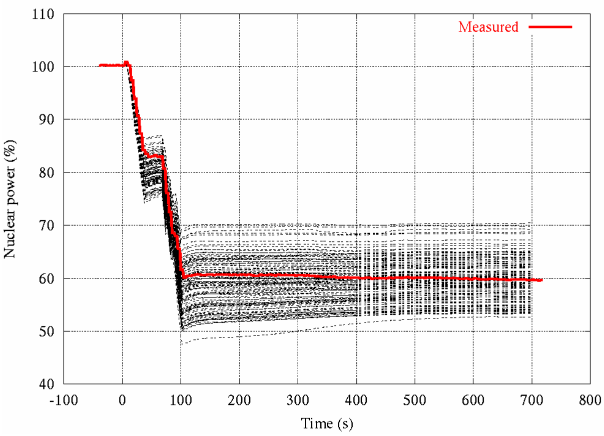
- Three-dimensional neutronics in rectangular fuel bundle geometry by TRAB
- Core and circuit hydraulics by SMABRE
- Fuel calculation by either code
- Data exchange inside each node in core
- Allows modelling of reversed flow in core and open core thermal hydraulics phenomena
- A stable and satisfactory steady state has been achieved
- Next step: EPR core modelling, supplementing SMABRE's BWR circuit model for dynamics



Contact person: Hanna Räty, VTT, hanna.raty@vtt.fi

Reactor physics & dynamics analysis
Uncertainty and sensitivity analysis

- Replacing conservative safety analyses with best-estimate analyses requires support from uncertainty and sensitivity analysis
- Uncertainty analysis method has been developed at VTT for analyses with coupled 3-D neutronics/thermal hydraulics codes
- Consists of several computer runs with randomly varied input parameters and statistically analyzed output factors
- Uncertainty analysis methodology study has also been made for steady state reactor physics



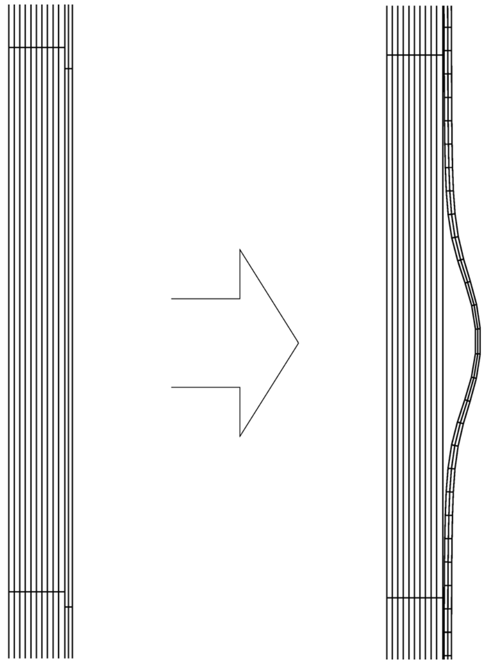
Measured neutron power in one turbine trip test in Loviisa-1 and 100 calculated results with the HEXTRAN-SMABRE code as a part of uncertainty and sensitivity analysis.

Contact person: Elina Syrjälahti, VTT, elina.syrjalahti@vtt.fi

New Mechanical Modelling

Finite Element Method based mechanical modelling in FRAPCON, FRAPTRAN and FRAPTRAN-GENFLO codes

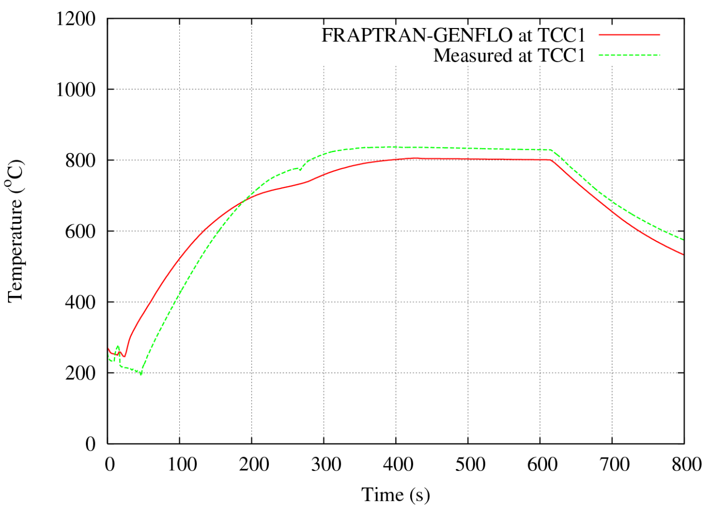
- Previous 1½-D thin-shell modelling replaced by FEM formulation
- Stress distributions within the cladding, large deformations, ballooning, pellet-to-clad friction
- Optional choise of detail in modelling



Contact person: Arttu Knuutila, VTT, arttu.knuutila@vtt.fi

FRAPTRAN-GENFLO Development

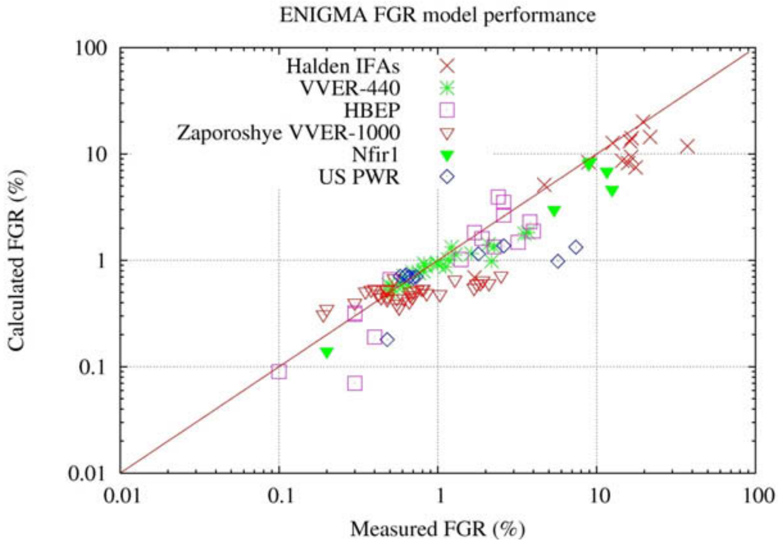
- FRAPTRAN fuel transient code coupled with advanced thermal hydraulics GENFLO
- Application to Halden IFA-650 LOCA experiment planning
- Nontypical geometry
- Heat transfer governed by radiation



Contact person: Jaakko Miettinen, VTT, jaakko.miettinen@vtt.fi

Re-correlation of the fission gas release model in ENIGMA code

- ENIGMA half-mechanical diffusion model; burnup-dependent parameters
- 108 datapoints from data libraries, power reactors, test reactors
- Automated least squares fit of seven parameters

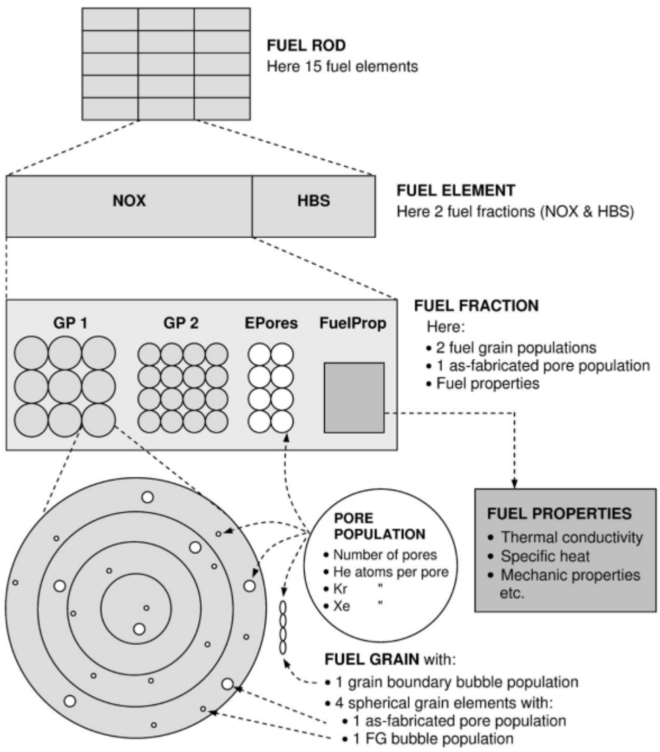


Contact person: Arttu Knuutila, VTT, arttu.knuutila@vtt.fi

Validation of a mechanistic fission gas release model

Coupling of an Object Oriented Programming module with VTT version of ENIGMA

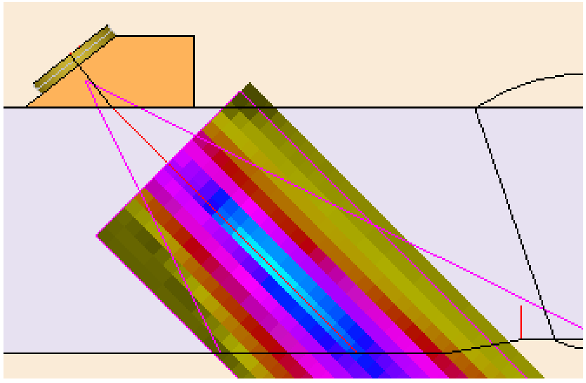
- Exercise to merge two programming cultures and model validation
- Successful operation of the two codes together
- Detailed mechanistic handling of fission gases
- Basis for future development

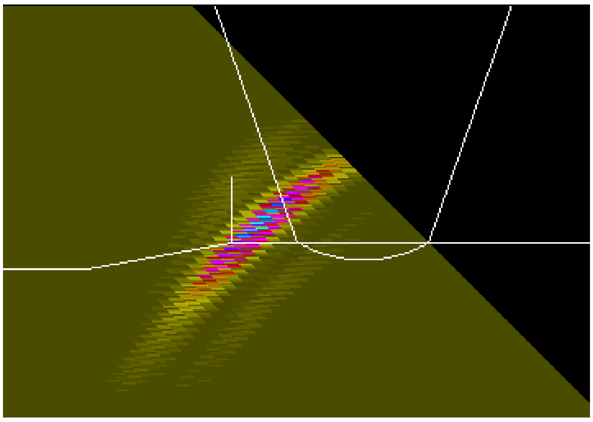


Contact person: Laura Kekkonen, VTT, laura.kekkonen@vtt.fi

Computer Simulation of Ultrasonic Testing

- Application of advanced modeling and simulation program codes to compute ultrasonic field of transducers.
- Computation of field/ reflector interaction to predict indications and their amplitudes in the practical inspection applications.
- Important and powerful tool for ultrasonic inspection design and qualification.
- Verification tests applying conventional transducers on reflectors with different tilt and skew angles produced satisfying results.

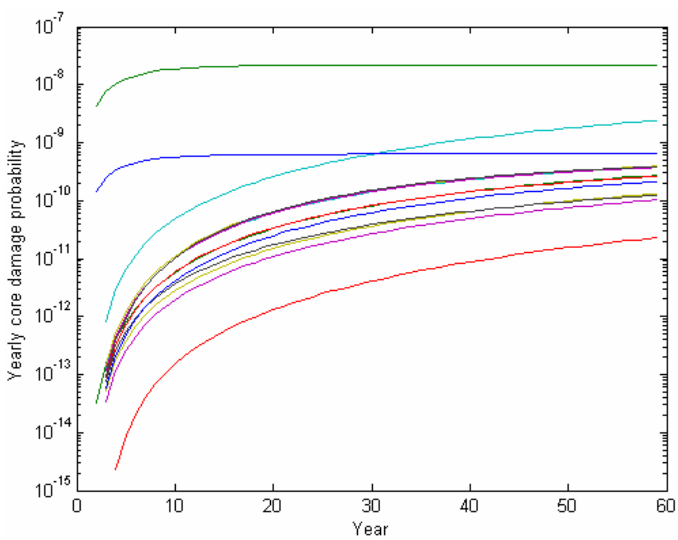




Contact person: Matti Sarkimo, VTT, matti.sarkimo@vtt.fi

Risk-informed in-service inspection

- Development + testing of a semi-quantitative version of EPRI RI-ISI risk matrix analysis approach
- Failure potential analyses: with fracture mechanics and Markov matrix based applications
- Risk analyses: with Markov system analysis application
- Pilot target: Shut-Down cooling system in a Finnish BWR.

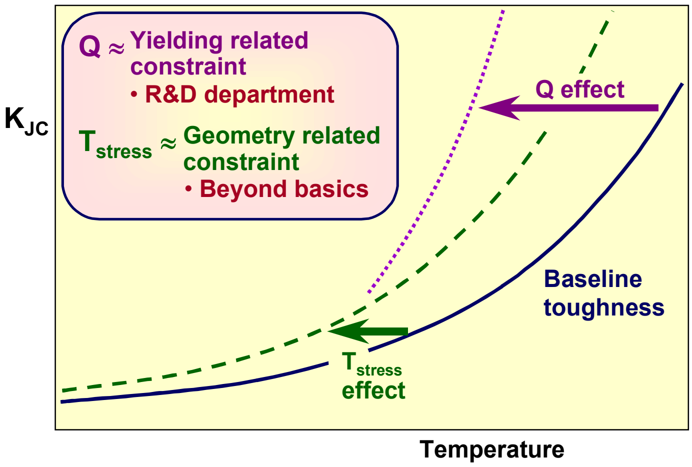


Yearly core damage probabilities for some piping welds.

Contact person: Otso Cronvall, VTT, otso.cronvall@vtt.fi

Transferability of test results for structural analysis

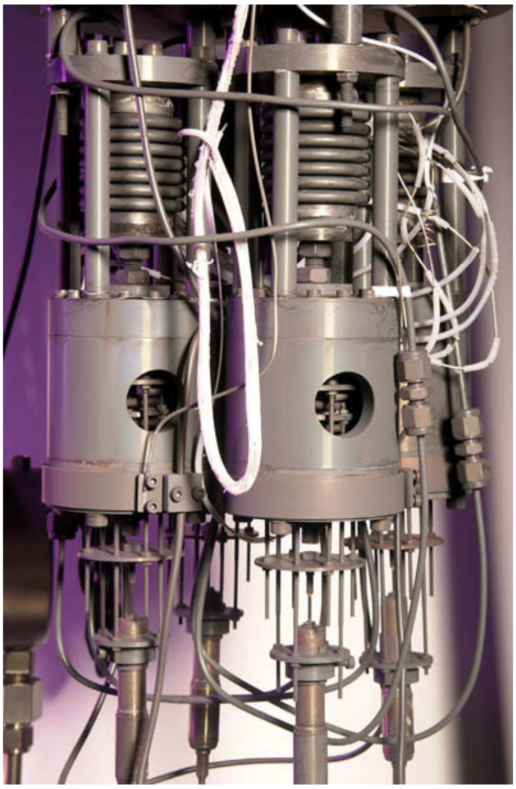
- Development of the Master Curve method and ASTM E1921 including standard-ization activities & small specimen testing techniques.
- Introduction of the so-called 'Bi-Modal' Master Curve for heterogeneous fracture mechanical behavior, such as in welds
- Experimental testing of different specimen and crack geometries in order to implement the Master Curve method with T-stress & Q fracture toughness



Contact person: Kim Wallin, VTT, kim.wallin@vtt.fi

Bellows Fatigue Unit

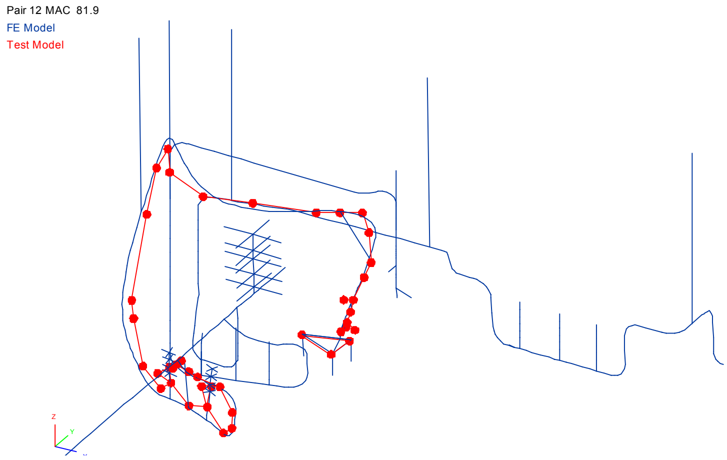
- Verified for LCF, HCF & spectrum straining in LWR hot water
  - alignment 2% dyn
  - direct strain control
  - dynamic load calibration 0.5 Hz
  - water chemistry in hot loop in situ testing in test reactor



Contact person: Jussi Solin, VTT, jussi.solin@vtt.fi

Piping Vibration

- FE-models with five different scales built and compared against measured data
- Force identification and forced vibration analysis performed with the most extensive model
- Measured operational displacement shapes compared with the measured natural mode shapes
- Assessment of operational stresses based on operational modes

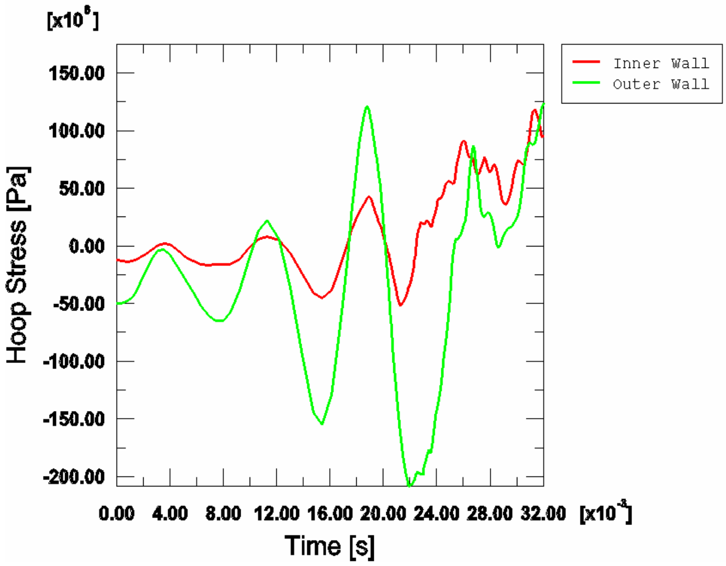


Comparison of measured (red) natural mode shape with its numerical counterpart (blue), predicted with the most extensive model containing also the adjacent lines.

Contact person: Ari Vepsä, VTT, ari.vepsa@vtt.fi

Analysis of BWR Suppression Pool Behavior

- CFD modelling of the phenomenon
- injection of air has been modelled by using the Volume of Fluid model
- water hammer induced a by large condensing steam bubble has been modelled with single-phase CFDcalculations
- homogenous two-phase model for simulations with condensation has been implemented and tested in Star-CD and Fluent



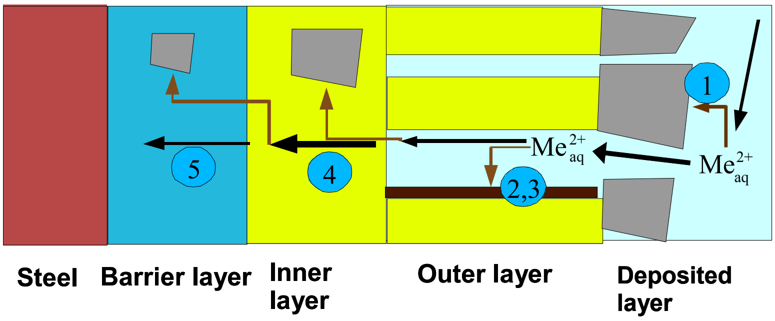
Single-phase simulation of a rapidly condensing steam bubble: hoop stress at bottom rounding.

Contact person: Antti Timperi, VTT, antti.timperi@vtt.fi

Further development of the Mixed-Conduction Model of oxide films in LWRs emphasising surface complexation and reprecipitation (LWROXI)

A 4-layer oxide model to describe the incorporation from NPP coolant to material surfaces

- describes the processes related to transport of species in different oxide film layers
- combines relevant high-temperature analytical experiments with theoretical modelling
- critical rate limiting steps in the incorporation processes can be distinguished



Contact person: Petri Kinnunen, VTT, petri.kinnunen@vtt.fi

CONTECH consists of several small practical projects dealing with concrete structures

Examples of topics that have been studied:

- internal curing agents of concrete,
- very rapid hardening grouts,
- maximum acceptable extent and type of rust on reinforcement bars during construction,
- cathodic protection of reinforcement,
- cumulation of salts in concrete,
- quality requirements and verification methods of concrete protective agents,
- non-destructive testing methods of reinforcement corrosion
- pretreatment of concrete surface for hydrophobic impregnation
- quality requirements and verification methods of concrete curing agents
- quality requirements of self-compacting concrete
- quality requirements of form liners and their usage
- effect of form liners and hydrophobic impregnation on chloride permeability

Contact person: Liisa Salparanta, VTT, liisa.salparanta@vtt.fi

Very rapid hardening mortars, grouts and concretes

The applicability of a very rapid hardening grout for grouting joints between concrete slabs and steel piles was studied in laboratory.

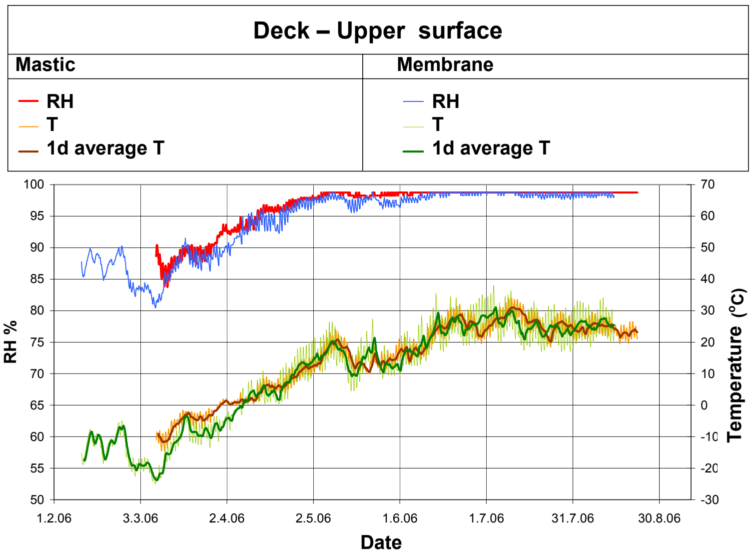
- Castability of the grout was good
- No bleeding was observed
- Vibration limit was on the safe side compared to that given by the manufacturer
- Heat generation was rapid in the beginning of the hardening
- The very rapid hardening grout tested is suitable for grouting joints between concrete slabs and steel piles

Contact person: Pertti Pitkänen, VTT, pertti.pitkanen@vtt.fi

RH-measurements using sensors cast in concrete

Sensors were installed on concrete bridges before casting concrete in 1995–1998. RH- and T -measurements were carried out in 1995–2001. The aim of the current study was to check if the sensors are working properly and to measure the present relative humidity of the structures. The duration of the measurements was half a year and the measurements were carried out during the coldest and warmest periods of the year.

- All of the sensors are working
- The humidity of the bridge deck with mastic waterproofing had not changed during the years
- The bridge deck with membrane waterproofing had dried

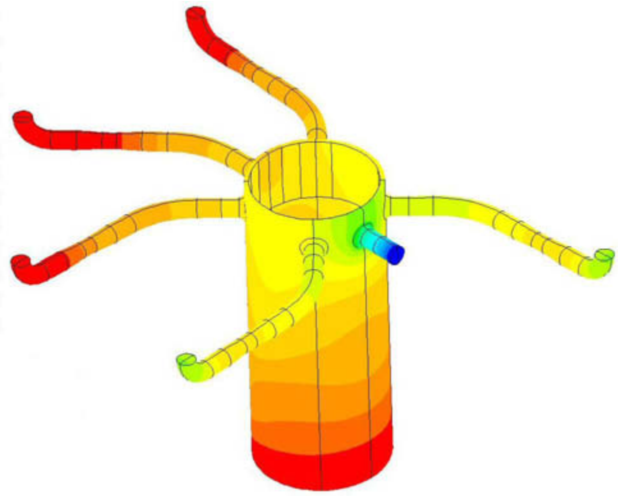


Contact person: Liisa Salparanta, VTT, liisa.salparanta@vtt.fi

MULTIPHYSICS

Fully coupled CFD – FEA Fluid Structure Interaction (FSI) calculations

- Fully coupled FSI calculations using the CFD codes FLUENT and Star-CD and FEA code ABAQUS.
- FSI analyses of the LBLOCA was also carried out entirely in ABAQUS using acoustic elements.
- Boundary conditions for the pipe break were calculated using the system codes APROS and TMOC.
- Both FSI and FEA results seem to be reasonable, but validation against experiments is the next step of the MULTIPHYCIS project.

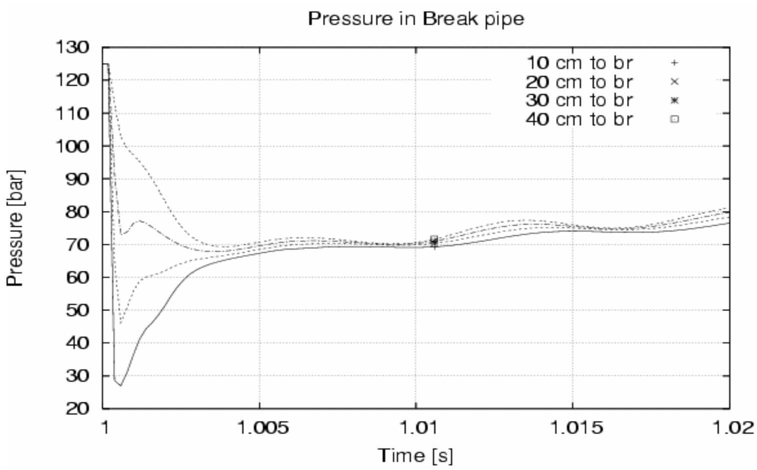


Contact person: Ville Lestinen, FNS, ville.lestinen@fortum.com

MULTIPHYSICS

Boundary conditions calculations with the system codes

- System codes APROS and TMOC were used to calculate the pressure drop in the pipe break.
- There were some differencies in the results of these two codes. The reason is still partly unclear, but analysis of the differencies is in progress.
- Also system code calculations will be validated in the next step of the project.

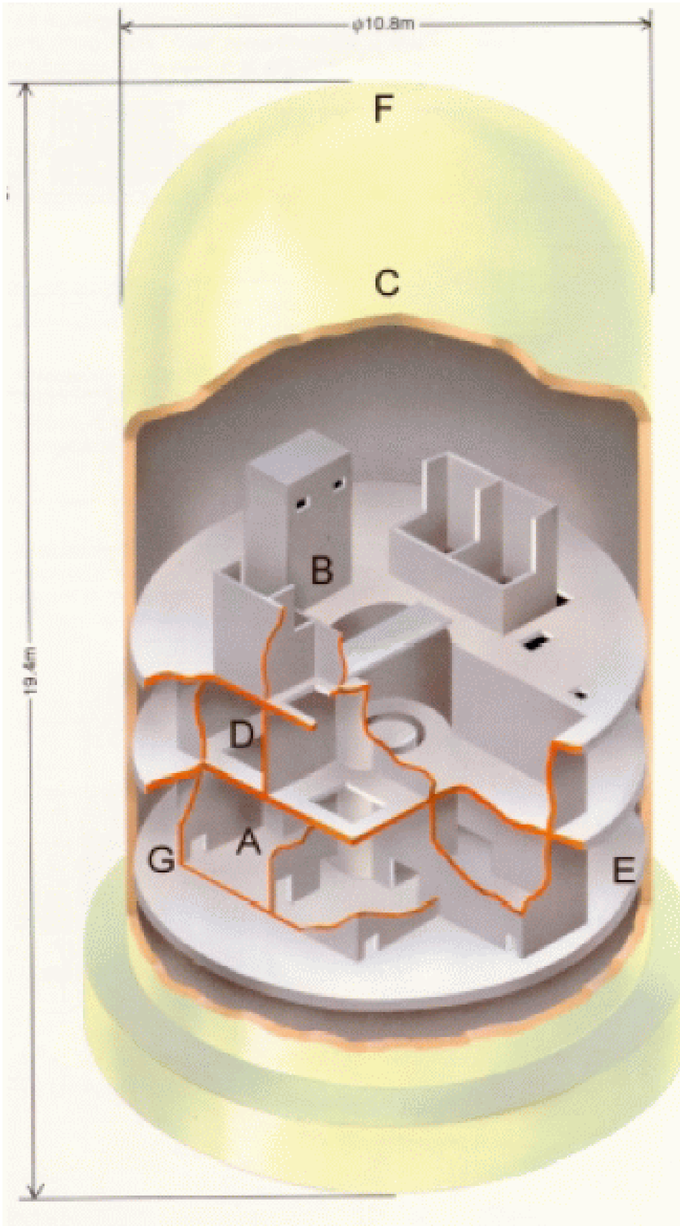


Contact person: Markku Hänninen, VTT, markku.hanninen@vtt.fi

Development of APROS containment model

The project of 2005 was related mainly on containment model development. The improvements included following topics.

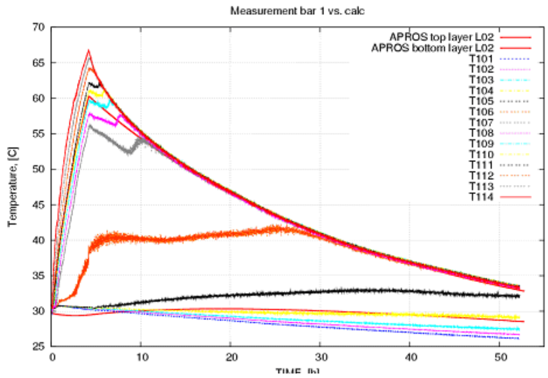
- Development of two new containment internal spray model and improvements in spray models. Validation of the old model and one of the new models against International Standard Problem No. 35 made in the Nupec facility illustrated in the figure.
- Diverse development topics for the APROS containment model in the project were development of new condensation and evaporation calculation for high steam concentration, development of water film model on containment structures, addition of valve components and development of new concentration calculation, which was added into APROS thermal hydraulic model too.



Contact person: Mika Harti, Fortum Nuclear Services, mika.harti@fortum.com

Development of APROS containment model (cont.)

- Boiling water reactor modelling improvements covered validation of water pool temperature stratification model (ment to be used to model BWR suppression pool) against Poolex experiments STB-20 and STB-21 (see figures), development of steam separator model for APROS thermal hydraulic model (with different parameters the model can be used to describe the steam dryer too) and testing of APROS thermal hydraulic model temperature statification capabilities in sea water channel conditions.



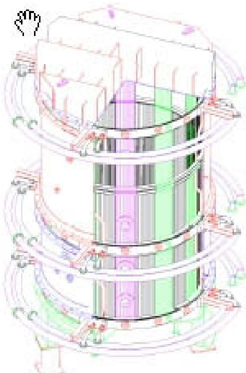
- New correlations for condensing heat exchangers in APROS thermal hydraulic model.

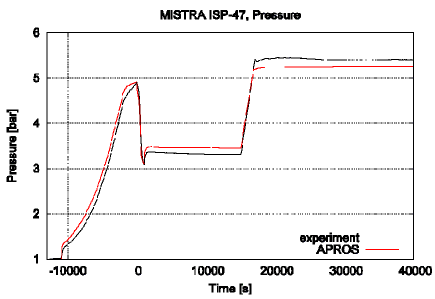
Contact person: Mika Harti, Fortum Nuclear Services, mika.harti@fortum.com

Validation of APROS containment model

The project of 2006 was related mainly on containment model validation against three experiments

- International Standard Problem No. 42 phase F.
- International Standard Problem No. 47 Mistra experiment. The Mistra experimental facility and preliminary results are presented in the figures.
- Pacos Px1.2 experiment.

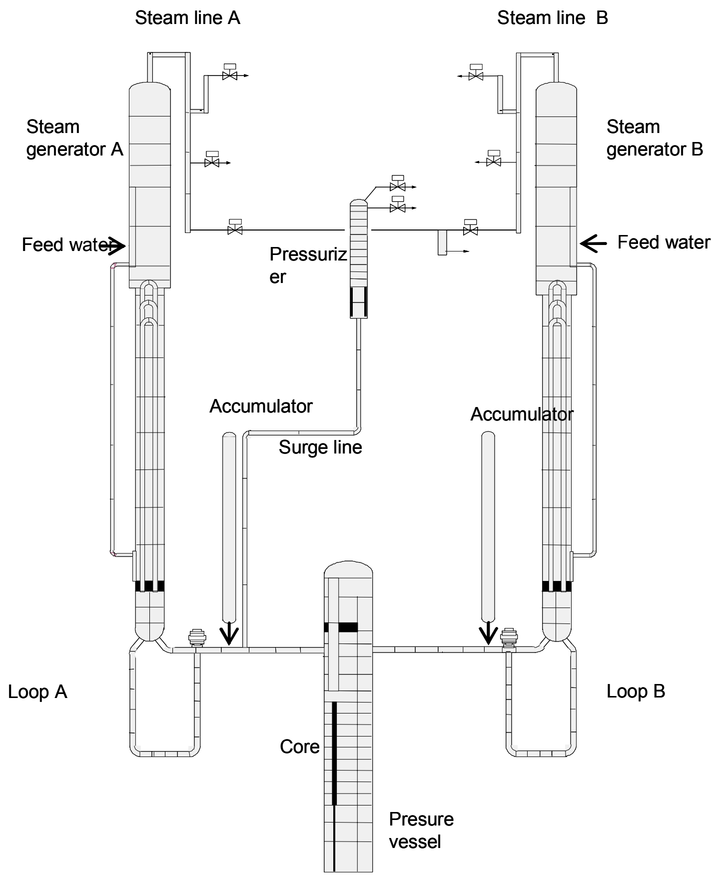




Contact person: Mika Harti, Fortum Nuclear Services, mika.harti@fortum.com

APROS validation

- PACTEL NCg
  - effect of non-condensable gas on heat transfer in horizontal steam generator
- PKL
  - midloop operation
  - boron concentration
- ROSA
  - SBLOCA in upper head
  - break up of the penetration nozzle of the control rod drive mechanism

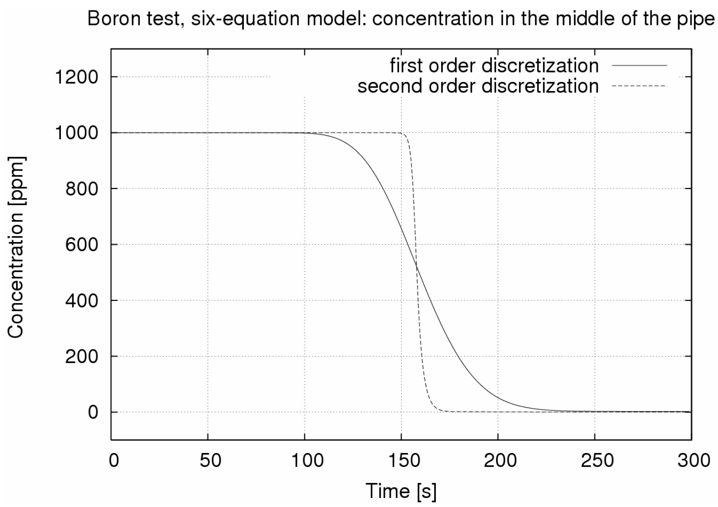


Contact person: Ismo Karppinen, VTT, ismo.karppinen@vtt.fi

Boron concentration tracking in APROS

Boron concentration affects reactivity and core power

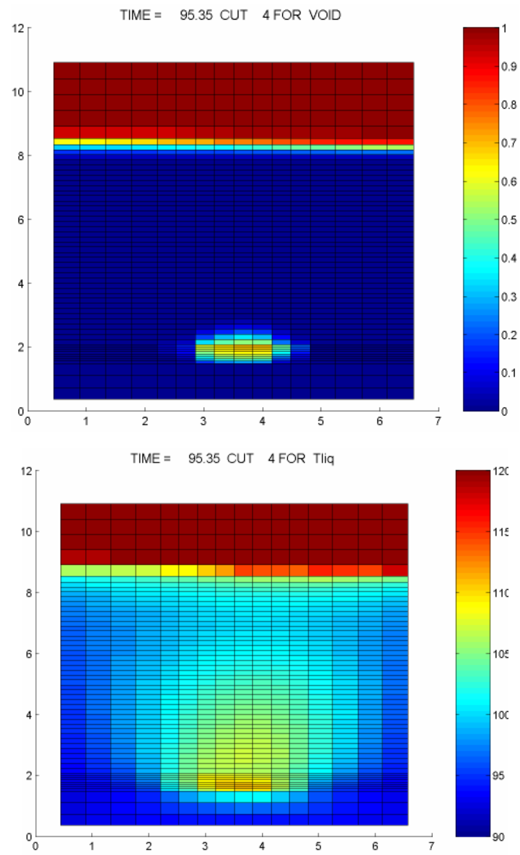
- Numerical diffusion with 1st order discretization
- 2nd order discretication implemented in APROS
- Better tracking of sharp concentration fronts



Contact person: Ismo Karppinen, VTT, ismo.karppinen@vtt.fi

Development of the 3D porous media code PORFLO

- Five-equation two-phase flow model
- Phase separation solved using drift-flux model
- SIMPLE (Semi-Implicit Method for Pressure-Linked Equations) pressure corrector methods
- Faster solution algorithm allows large meshes

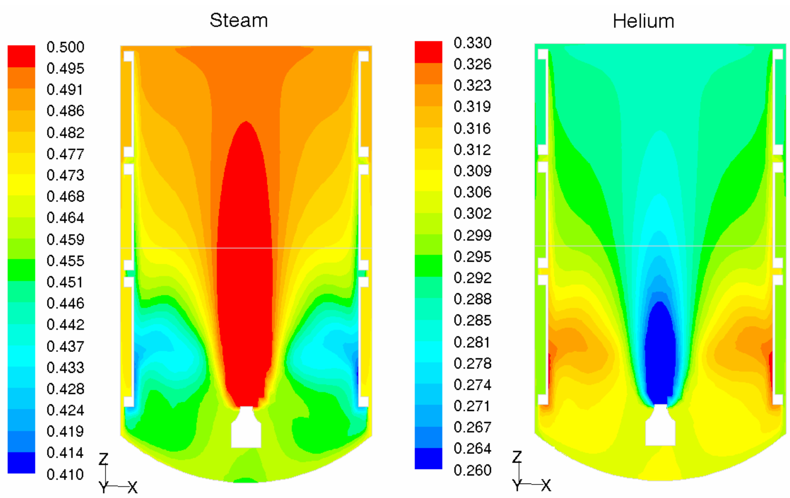


Void fraction (on the top picture) and temperature (on the bottom) distribution in the isolation condenser pool during initial phase of boiling calculated with PORFLO.

Contact person: Jaakko Miettinen, VTT, jaakko.miettinen@vtt.fi

Wall condensation in CFD code Fluent

- Testing of wall condensation model implemented at VTT
- Steam condensation in presence of air and helium
- MISTRA experiments in SARnet (Severe accident research network) co-operation

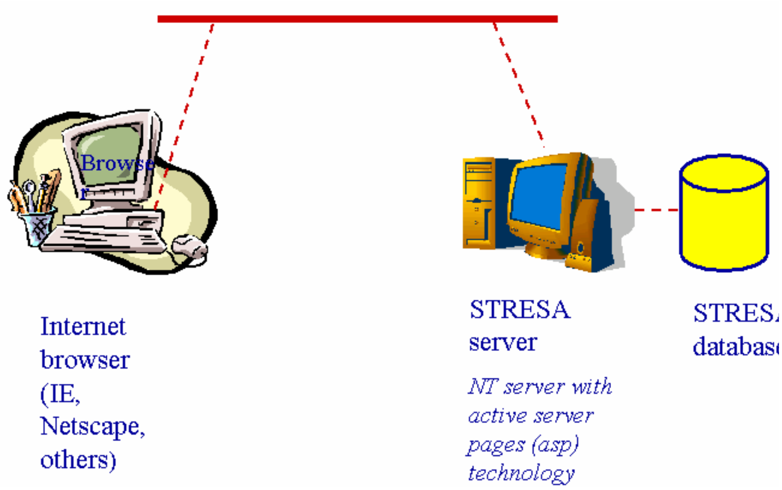


Volume fraction of steam and helium calculated with Fluent.

Contact person: Risto Huhtanen, VTT, risto.huhtanen@vtt.fi

Wall condensation in CFD code Fluent

- At Lappeenranta University of Technology several hundreds of thermal-hydraulic experiments have been performed since 1975.



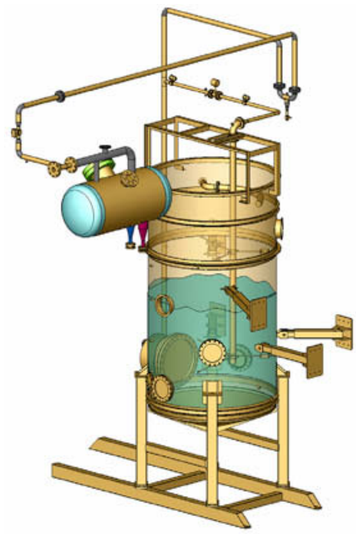
- Some of the data and documents were in danger to be lost because of uncompatible hardware and software in use today.
- Data and documents have been archived in the STRESA database and in CD and DVD disks. The total amount of the experiments in the database is almost 900.
- The archived data and documents can be used for code validation, for planning and understanding the future experiments, for educational purposes, and as a Finnish contribution in international co-operation projects.

Contact person: Vesa Riikonen, LUT, vesa.riikonen@lut.fi

Condensation Pool Test Facility

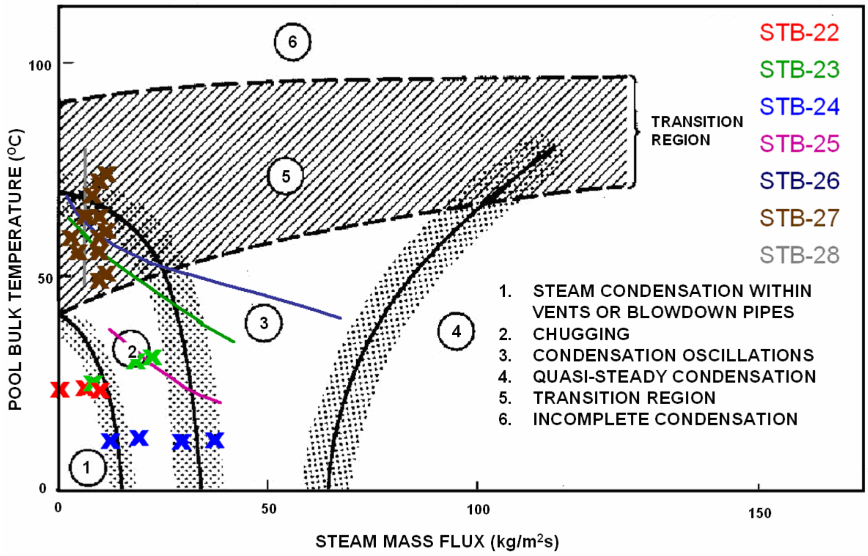
An intermediate scale experimental set-up enabling studies of different phenomena taking place in a BWR suppression pool during steam/gas discharge

- Large open water pool with a submerged vent pipe
- Windows for visual observation
- Wide frequency band instrumentation and data acquisition plus high-speed video equipment for capturing phenomena connected to rapid condensation



Contact person: Markku Puustinen, LUT, markku.puustinen@lut.fi

Condensation Pool Experiments

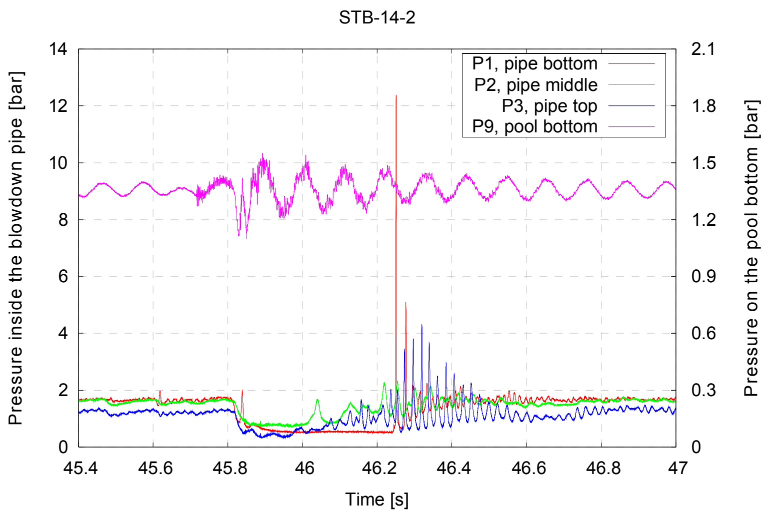


- Several regions of the condensation mode map of Lahey and Moody covered
- Experiments on bubble dynamics, pressure oscillations, chugging and stratification
- An extensive database generated for developing and assessing computing methods used for nuclear safety analysis

Contact person: Markku Puustinen, LUT, markku.puustinen@lut.fi

Pressure oscillations in the blowdown pipe

- Steam bubble at the blowdown pipe outlet condenses rapidly and a partial vacuum is generated
- Steam-water interface moves up inside the blowdown pipe
- Condensation-induced water hammer is initiated as the pipe is filled with water
- High pressure pulses are measured inside the blowdown pipe

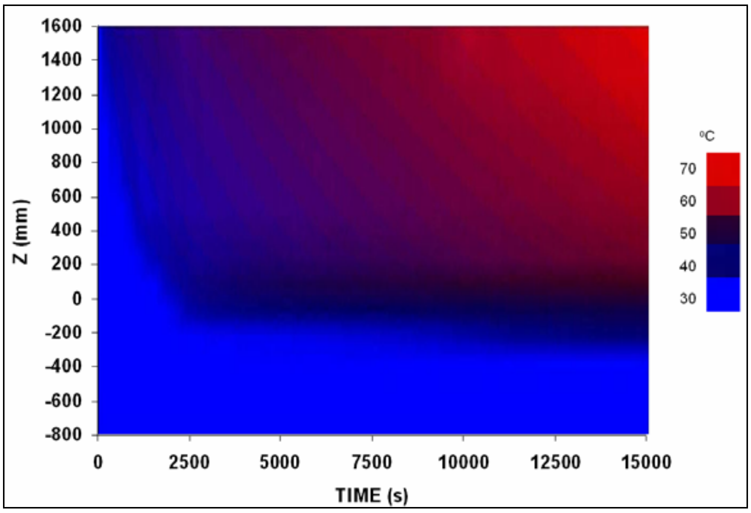


Contact person: Markku Puustinen, LUT, markku.puustinen@lut.fi

Thermal stratification of pool water

Experiment series on temperature stratification of the condensation pool

- Related to the later phase of a steam line break accident inside the containment
- Strong thermal stratification in vertical direction is developed above the blowdown pipe outlet elevation
- No mixing effects due to small steam flow
- Results used for the validation of the APROS stratification model

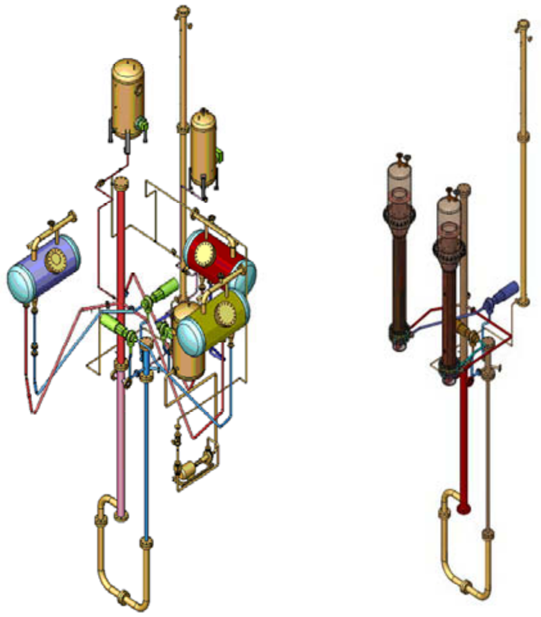


Contact person: Markku Puustinen, LUT, markku.puustinen@lut.fi

PACTEL for PWR studies

The PACO project was introduced to initiate and plan a PACTEL facility related project proposal for OECD, including experimental project planning and pre-calculations. Present scheme is to deliver the final proposal after a specialist meeting on the subject.

- A proposal to OECD on PACTEL related research program was prepared. The supporting pre-calculations were made with APROS
- A new set-up of PACTEL has been introduced as well as corresponding simulation models
- The upgrades to APROS by VTT developers have improved simulation performance

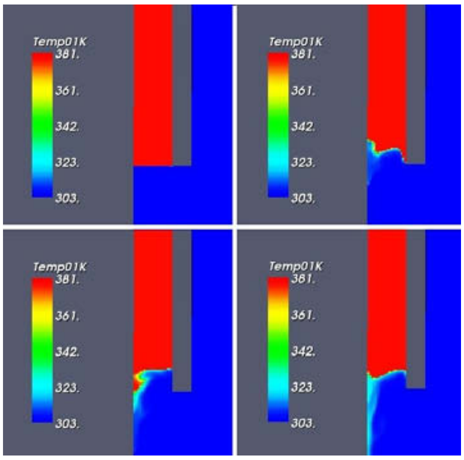


Contact person: Heikki Purhonen, LUT, heikki.purhonen@lut.fi

CFD validation at LUT

Lappeenranta University of Technology and VTT participates in the Nuclear Reactor Simulations (NURESIM) EU Project. The goal of the work is to take part in the development of the new Common European Standard Software Platform for modeling of the two-phase flow simulations of nuclear reactors.

- The NEPTUNE CFD module has been installed at LUT. Different calculational grids have been developed and tested. Simulations started with 2D geometry
- A tailored experiment was carried out in POOLEX project and is used for development of NEPTUNE CFD in SALOME platform
- The POOLEX experiment is being simulated in 3D geometry. Air as 3rd phase has caused problems. Validation work continues during a research visit in CEA Grenoble

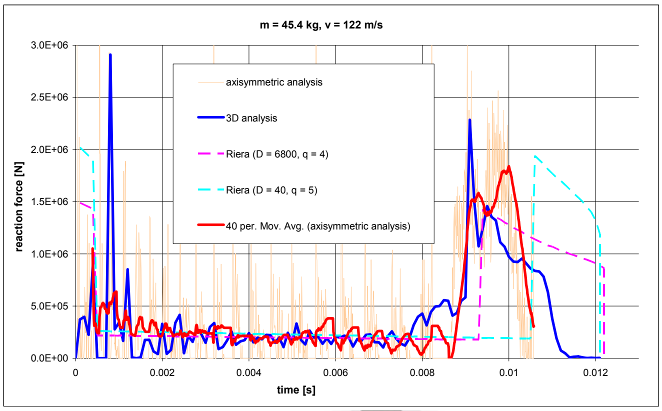


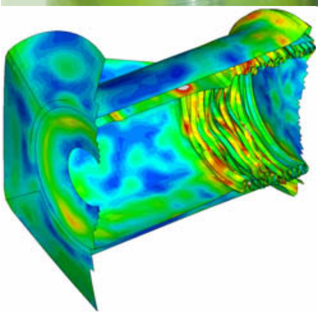
Contact person: Heikki Purhonen, LUT, heikki.purhonen@lut.fi

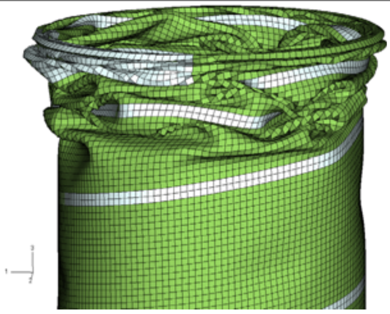
Wall Response to Soft Impact

Co-operation with Tampere University of Technology
Impact force-time functions for different types of missiles

- Riera method, finite element method (FEM)
- Numerical results verified against experimental data



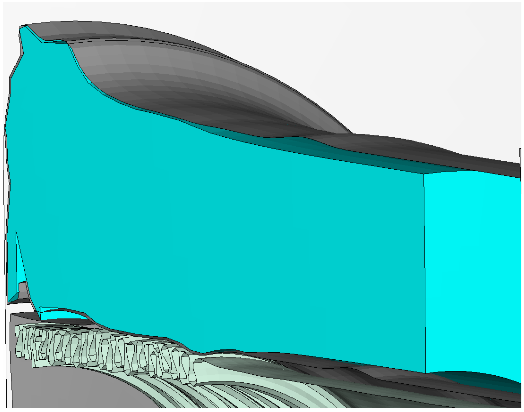


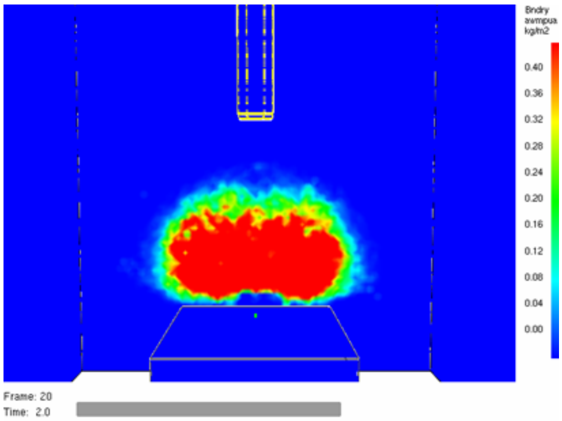


Contact person: Arja Saarenheimo, VTT, arja.saarenheimo@vtt.fi

Fuel dispersion in impact

- Fuel release and dispersal from impacting projectiles
- Drop size, liquid front velocity and extent of dispersal based on impact tests
- Simulation of fuel dispersal with Fire Dynamic Simulator (FDS) code

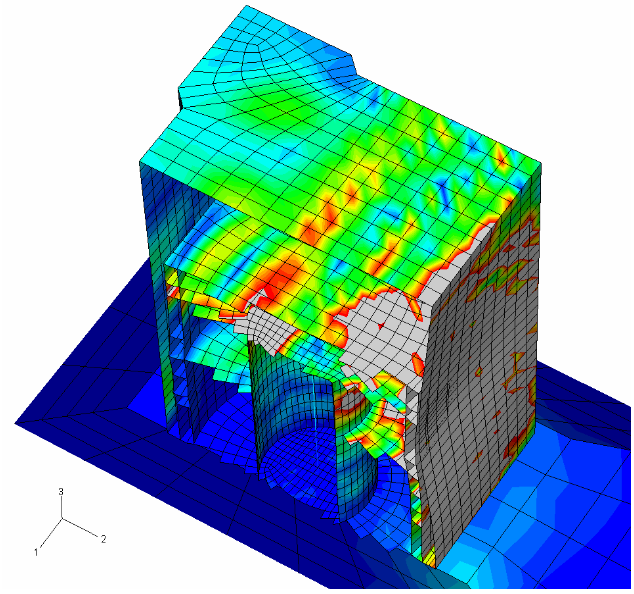




Contact persons:
Ari Silde, VTT, ari.silde@vtt.fi and
Simo Hostikka, VTT, simo.hostikka@vtt.fi

Structural response

- Calculation of force-time function due to an aircraft crash
- Structural integrity of a reactor building subjected to an aircraft impact
- Nonlinear reinforced concrete wall analysis tools verified against experimental data
- Floor response spectra due to excited vibrations

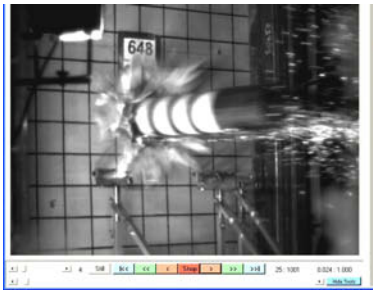


Contact persons:
Arja Saarenheimo, VTT, arja.saarenheimo@vtt.fi and
Kim Calonius, VTT, kim.calonius@vtt.fi

IMPACT Testing Facility

IMPACT test facility has been designed to study and measure the impact forces that would arise when an aircraft crashes against a structure or nuclear power station. The target has been a concrete wall with rebars and the missile a steel or aluminium pipe.

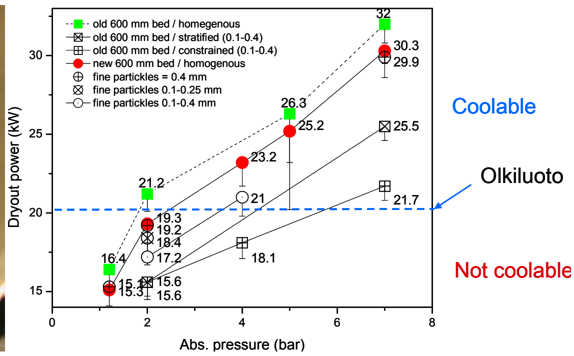
The main characteristics of the testing facility are:



- The mass of the missile is variable up to 100 kg and velocity can be accurately controllable between 100–200 m/s
- The wall must is reinforced concrete wall up to 40–60 cm thick
- Instrumentation includes high-speed cameras, force-time measurements, wall displacement measurements and rebar strains

Contact person: Ilkka Hakola, VTT, ilkka.hakola@vtt.fi

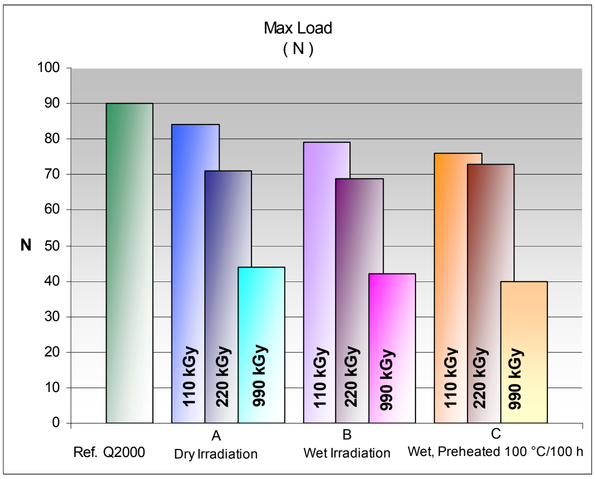
Coolability of Heat Generating Particle Beds



- STYX facility was constructed for measurements of dry-out heat fluxes in a particle bed
- Measured dry-out heat fluxes in homogeneous beds were lower than predicted with existing correlation models
- Measured dry-out heat fluxes were higher than predicted by existing correlation models for stratified beds with a layer of smaller particles on top of a coarser bed

Contact person: Ilona Lindholm, VTT, ilona.lindholm@vtt.fi

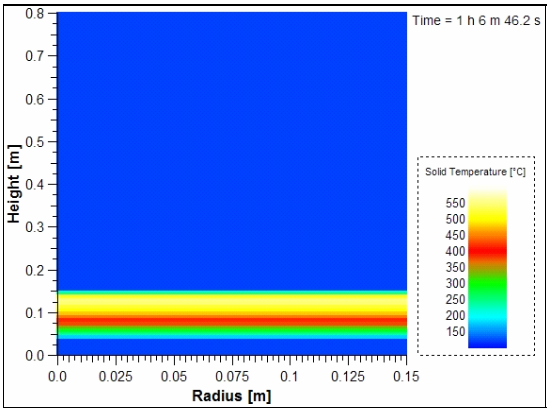
Testing of seal materials of containment penetrations

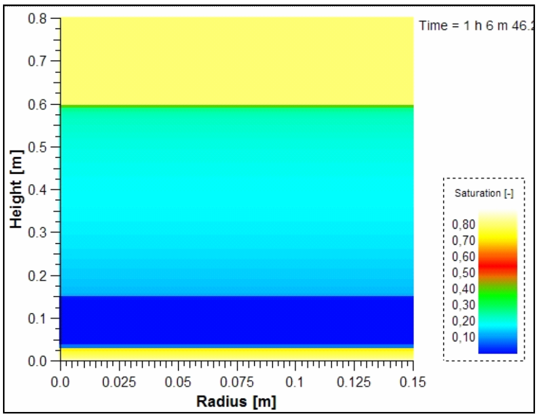


- Material properties of elastomers used as seals in nuclear power plants in long term under cumulative dose equivalent to 1.5 months, 4.5 months and 12 months after a severe accident
- Butyl Rubber survived worst and became soft and sticky, EPDM survived best but became harder, silicon became brittle
- Consequently butyl rubber seals were changed into EPDM seal in Olkiluoto plants

Contact person: Riitta Zilliacus, VTT, riitta.zilliacus@vtt.fi

Particle bed dry-out heat flux simulations with WABE code

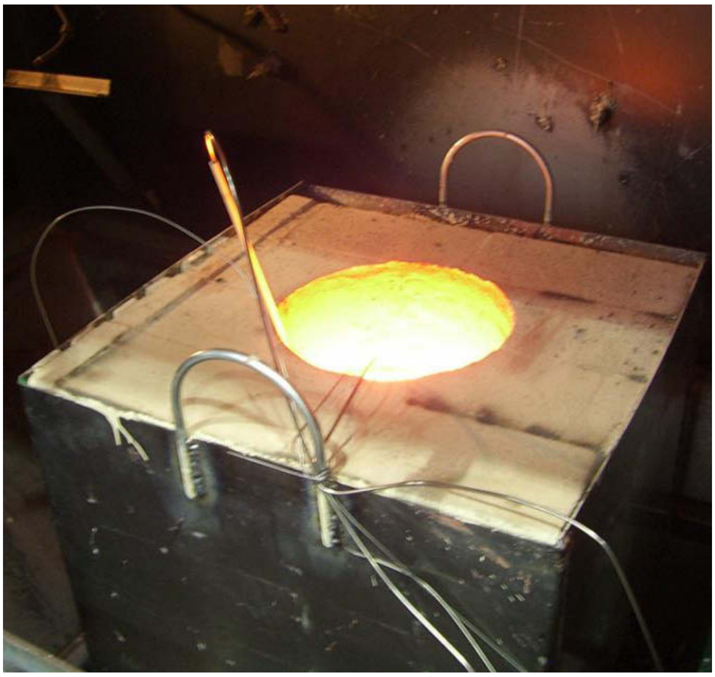




- WABE code developed at IKE Stuttgart was acquired for detailed analysis of the STYX tests and plant applications for Olkiluoto 1 and 2.
- Figure illustrates a calculated result of a STYX test 2.4 with dry region starting at elevation 10–12 cm from the bed bottom. In experiments dry zone was measured at 4–10 cm.
- Saturation value zero indicates totally dry region

Contact person: Jaakko Miettinen, VTT, jaakko.miettinen@vtt.fi

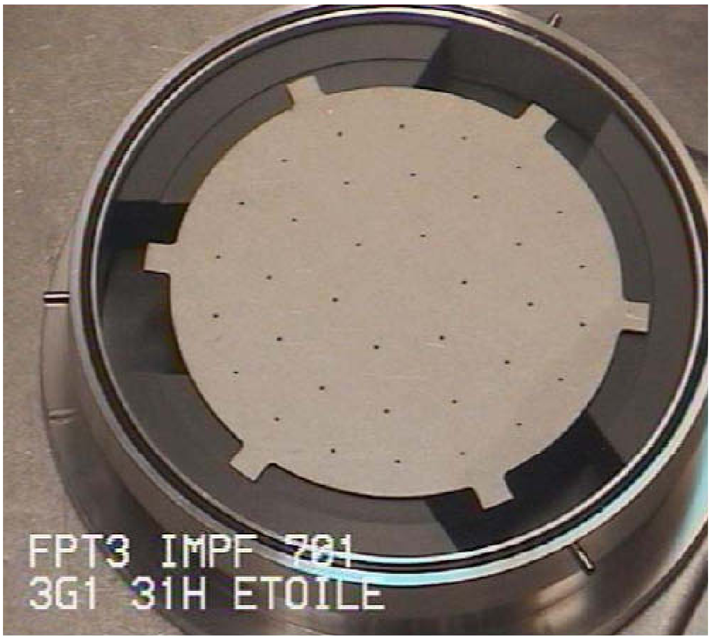
HECLA facility for investigation of metallic core melt-concrete interaction



- Experimental investigations with molten, superheated steel (T = 1900 °C) with ironoxide and siliceous concrete on concrete erosion mechanically and by melting
- Data applicable for material employed in melt collector cavity in Olkiluoto 3 plant
- Test facility constructed and scoping test performed
- Final experiments to be carried out in COMESTA project (2007–2008)

Contact person: Tuomo Sevón, VTT, tuomo.sevon@vtt.fi

Phebus FP experiments on severe accident phenomena



- Aerosol size distribution was successfully measured in Phebus FPT-3 experiment.
- Fission product revaporisation experiments using samples from Phebus FP has been conducted.
- Complicated deposition in steam generator has been solved greatly influencing mass balance of the fission products.
- Instrumentation for iodine measurements has been significantly improved.

Contact person: Ari Auvinen, VTT, ari.auvinen@vtt.fi

Ruthenium transport in primary circuit

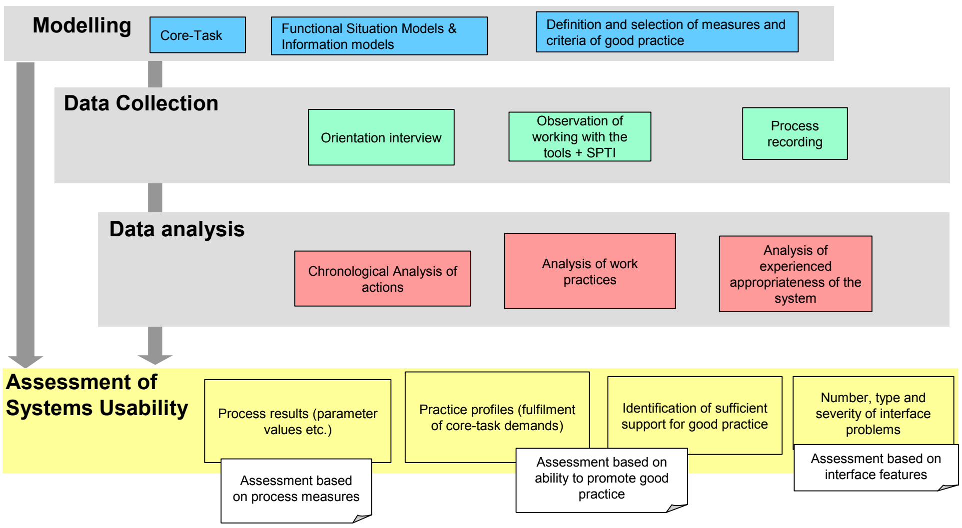
An experimental database on ruthenium transport in an air ingress accident was built up.

- With the help of sophisticated online γ-tracer measurement the role of different ruthenium oxides on ruthenium transport was determined.
- In addition to ruthenium transport experiments, ruthenium release from the fuel and behaviour in the containment were studied in the frame of Nordic NKS-R programme and European SARNET network.
- Within the study one scientific publication, one doctoral thesis and 11 presentations in international conferences have been completed. Two scientific publications are in preparation.

Contact person: Ari Auvinen, VTT, ari.auvinen@vtt.fi

Ruthenium transport in primary circuit

A performance-based method for integrated validation of control room human-system interfaces was developed. The method includes four phases.



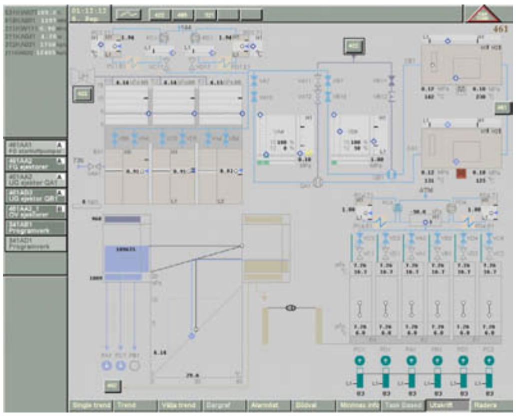
In systems usability the quality of the control room interface is evaluated against the tools's capability to promote the core task of process control and to fulfil key functions of a tool in practice. Hence the interface should be

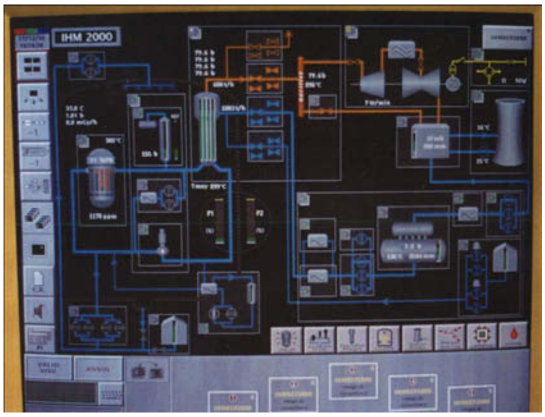
- effective and efficient in process control
- fit for human use and promote seamless human-technology and team collaboration
- meaningful for mastery of work and promote shared awareness

Contact person: Leena Norros, VTT, leena.norros@vtt.fi

Experimental testing of interface designs

Getting acquainted with and testing of different design concepts, i.e. Function Oriented Displays, Ecological Information Displays and presently available advanced displays. Study was a collaboration with HRP, EdF and University of Toronto.



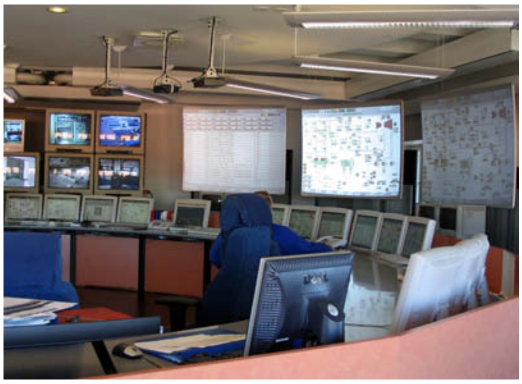


The expected result of the on-going studies are a better understanding of the pros and cons of the different approaches for operator work. Results provide guidelines for an integrated interface design concept to be developed for future control rooms.

Contact person: Leena Norros, VTT, leena.norros@vtt.fi

Operator experiences on working in screen-based control rooms

The project focused on gathering of operators' experiences regarding working in digitalized screen-based control rooms.



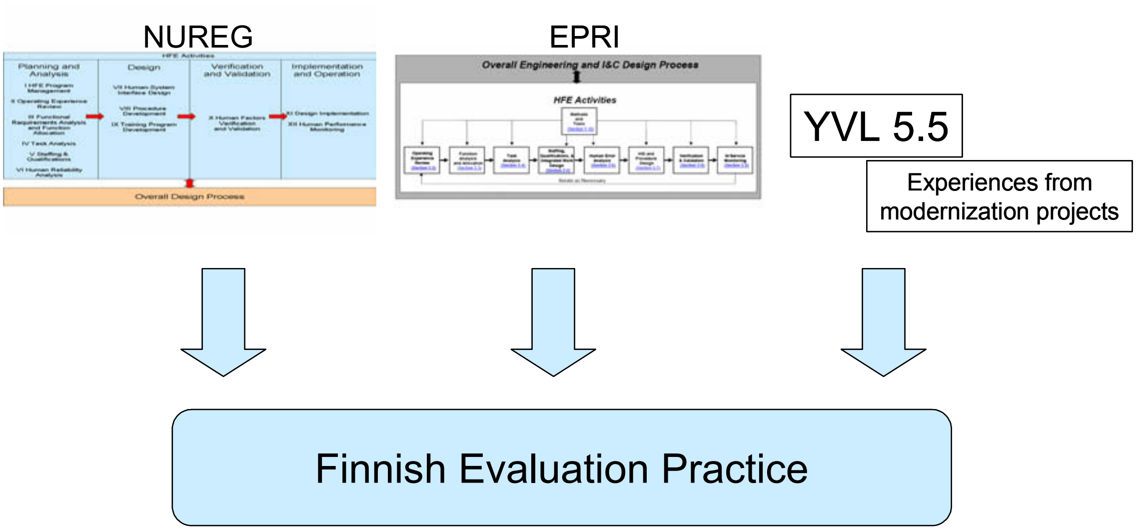
Results indicate that

- The amount of available process information has increased, providing faster diagnosing of failures
- Flexibility and the possibilities to tailor the new user interface are appreciated
- The role and practices of using large screen displays are not yet very clear
- ICT allows for more flexible roles and responsibilities, putting into practice fails
- Introduction of desktop-based workstations change communication and collaboration between operators; the issue should be addressed during training
- Development of trust in the system.

Contact person: Leena Salo, VTT, leena.salo@vtt.fi

Management of design processes from human factors engineering (HFE) point of view

HFE practices in the Finnish control room modernisation projects were studied. Practices were compared with international guidelines. As a conclusion a Finnish adaptation of the international HFE management guidelines is proposed.

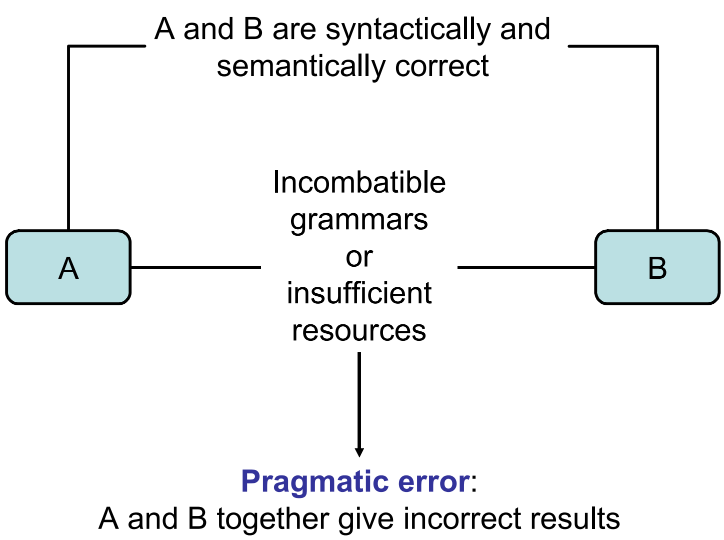


Contact person: Jari Laarni, VTT, jari.laarni@vtt.fi

Efficiency for Software Qualification

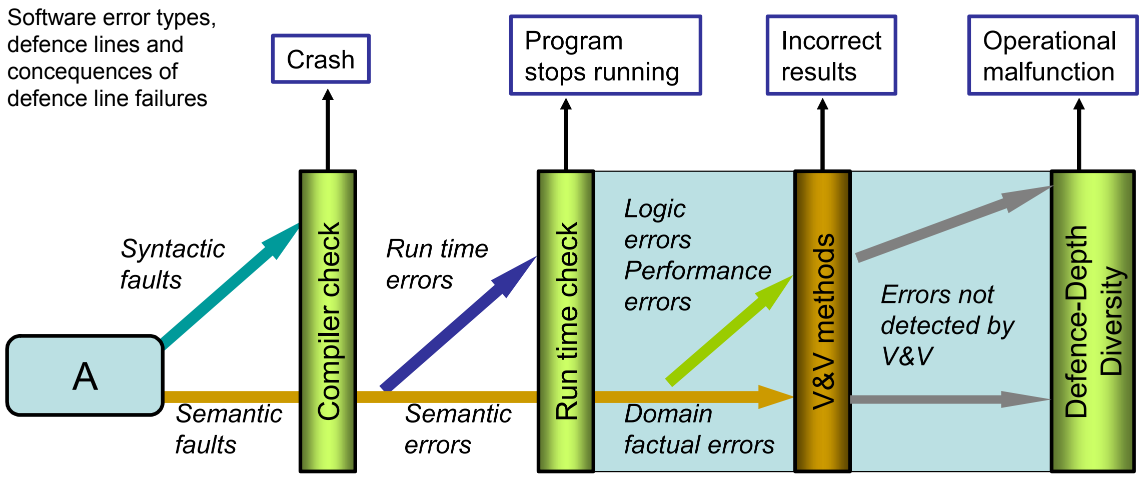
The developed new approach supports the demonstration of correctness of safety critical software in instrumentation and control systems. The approach is based on the three concepts of linguistics.

- Demonstrated at various development phases and at common cause failure analysing phase.
- Presented various qualification styles that utilize the new approach.
- Validated by a number of incidents involving software errors.



Contact person: Hannu Harju, VTT, hannu.harju@vtt.fi

Efficiency for Software Qualification



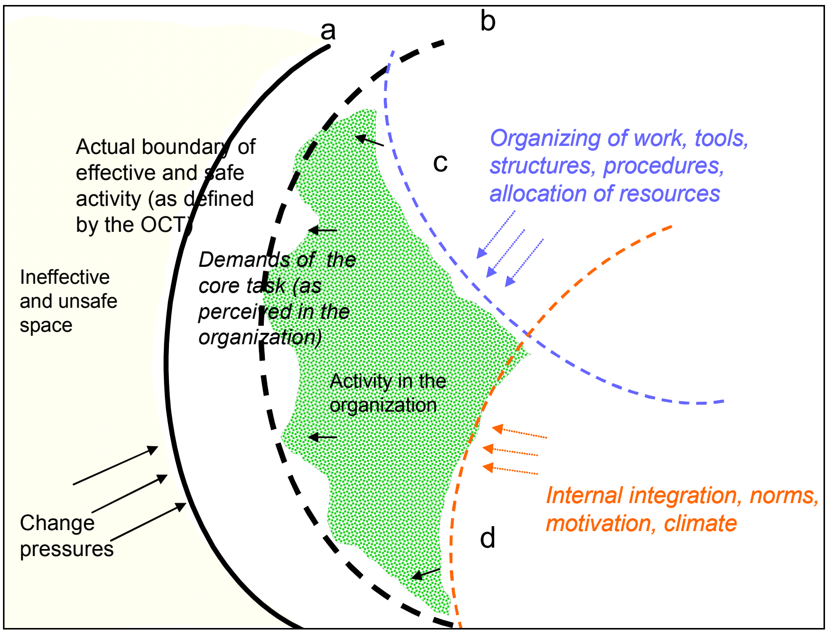
- For software with a single component, syntactic faults and run time errors can be detected reliably, but logic, performance and factual errors with great difficulty.
- For software with multiple components/ projects/ organisations, pragmatic errors take effect – they are simple errors, but difficult to detect and prevent.

Contact person: Hannu Harju, VTT, hannu.harju@vtt.fi

Model of organizational factors and safety

Organizational drift and local optimizing of practices

- Describes organizational drift of practices and norms toward unsafe condition
- Describes the elements of the organizational (safety) culture that influence actions
- Based on case studies in maintenance and engineering and international safety research



Contact person: Teemu Reiman, VTT, teemu.reiman@vtt.fi

Organizational challenges of safety critical organizations

- Book on the characteristics of special challenges of safety critical organizations (both in Finnish and English)
- Conceptualization of the organizational and psychological issues that the organizations must solve and clarification of the safety management principles
- Based on case studies, literature review, and benchmarking in e.g. aviation, chemical and oil industries

Contact person: Pia Oedewald, VTT, pia.oedewald@vtt.fi

Organizational challenges of safety critical organizations

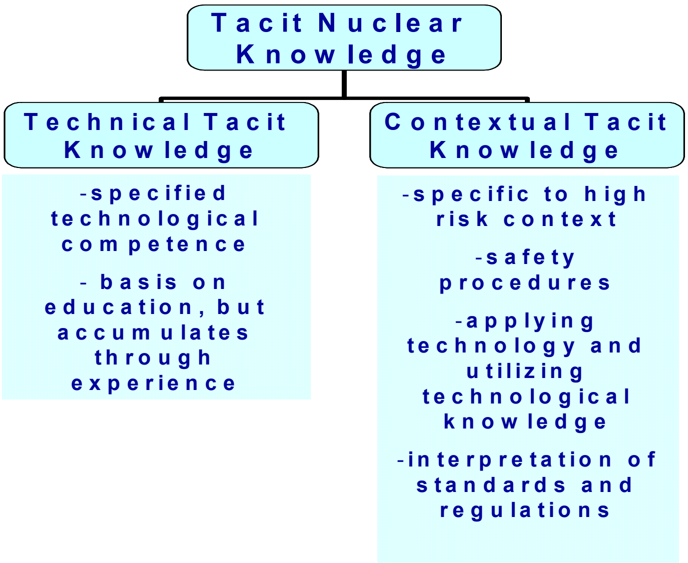
- Multiple measures of organizational culture in NPP maintenance units
- Conceptualization of the maintenance personnels' views on safe and effective maintenance work
- Identification of the challenges of maintaining safety culture in maintenance and a model of the demands of the maintenance core task

Contact person: Teemu Reiman, VTT, teemu.reiman@vtt.fi

Tacit Nuclear Knowledge

The role of tacit knowledge in the NPP context was specified. It was considered critical for three reasons: 1) nuclear technology is complex, 2) nuclear know-how is only in hands of a few and 3) safety and quality of plant operation are essential.

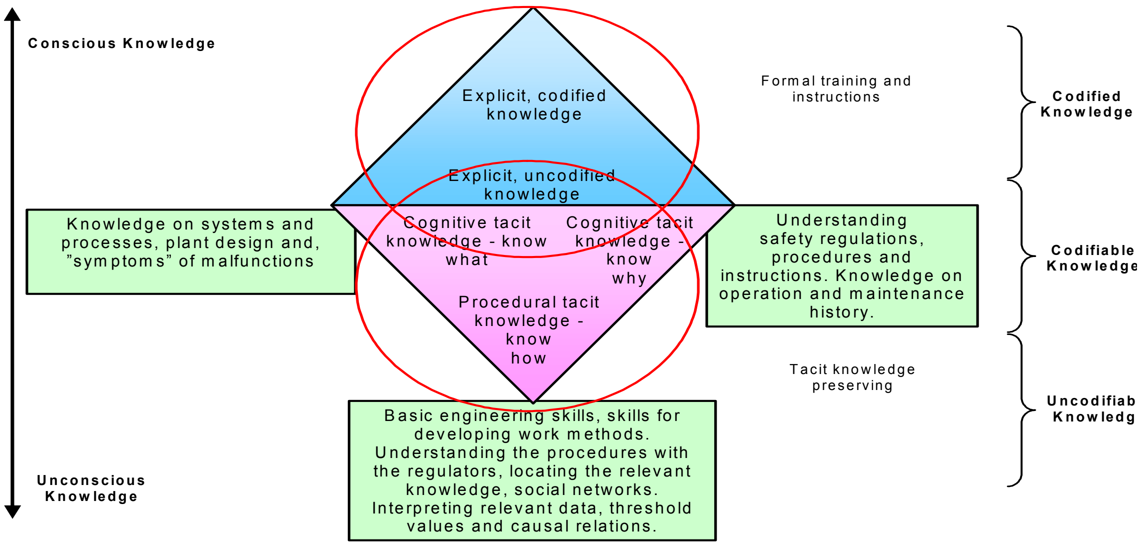
- The consept of tacit nuclear knowledge was defined. Tacit nuclear knowledge consists of two dimensions: technical and contextual tacit knowledge



Contact person: Tanja Kuronen-Mattila, HUT, tanja.mattila@tkk.fi

The DIAMOND Model

The DIAMOND model describes the nature and content of tacit nuclear knowledge in maintenance, technical design and control room. It is based on case studies and tacit knowledge literature.



Contact person: Tanja Kuronen-Mattila, HUT, tanja.mattila@tkk.fi

Challenges in and Prerequisites for Sharing Tacit Knowledge

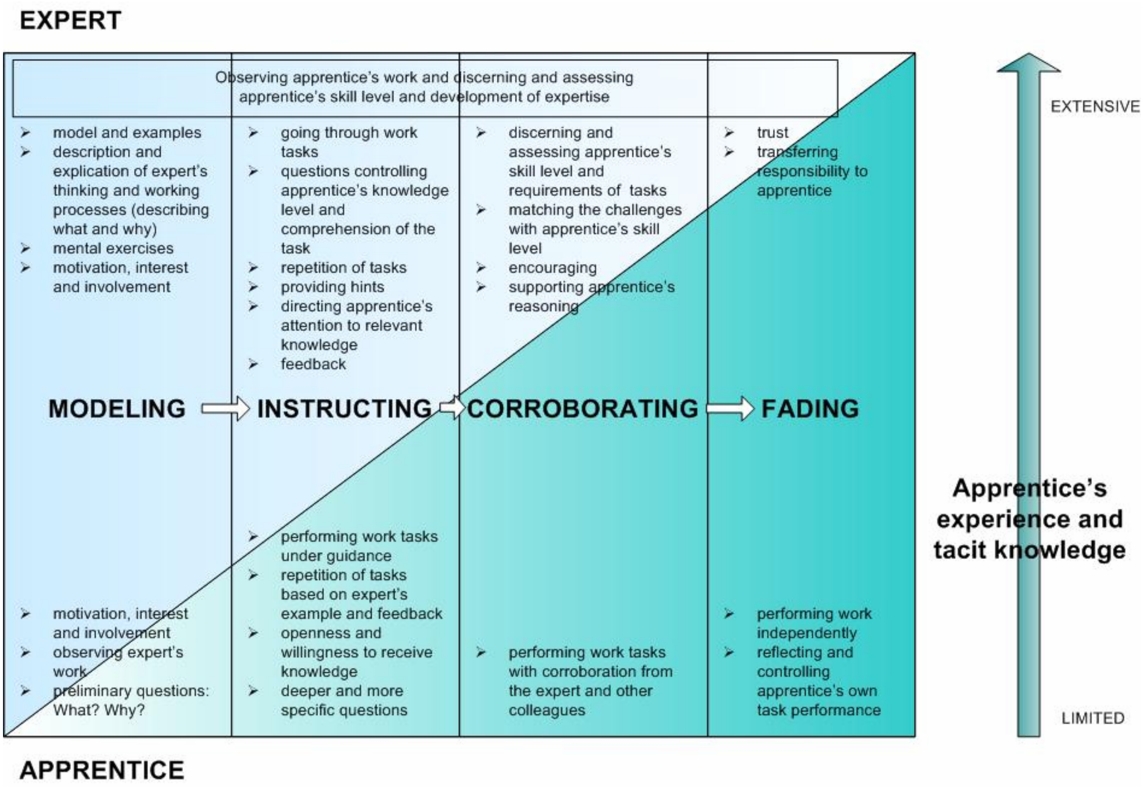
The challenges in preserving tacit knowledge in NPPs were recognized: 1) insufficient understanding of tacit knowledge 2) retirements of experienced experts 3) lack of opportunities for collaboration and 4) lack of competent instructors.

Prerequisites for sharing tacit knowledge in NPP's were identified:

Organizational factors	Situational factors	Social factors
Identifying experts	Authentic context	Collaboration
Early recruitment	Supportive atmosphere	Interpersonal relations
Resources	Communication and feedback	Individual qualities

Contact person: Tanja Kuronen-Mattila, HUT, tanja.mattila@tkk.fi

Apprenticeship as a Method for Preserving Tacit Knowledge



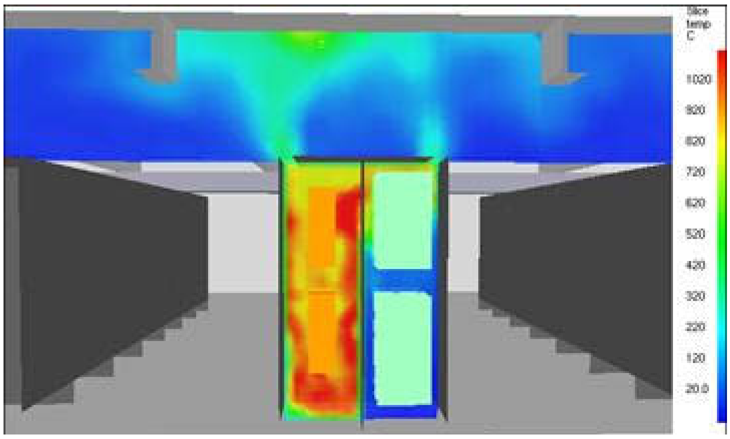
Good practices for preserving tacit knowledge in studied contexts were identified. These together with the analysis of literature produced a process model of apprenticeship. It is utilized in e.g. operator training.

Contact person: Tanja Kuronen-Mattila, HUT, tanja.mattila@tkk.fi

Two Model Monte Carlo tool

A two model Monte Carlo tool has been developed to run Monte Carlo using CFD fire simulations needed in large NPP compartments.

- Monte Carlo needs some 1000 fire simulation runs
- Too expensive for fire-PSA work using CFD
- Two model Monte Carlo needs some 30 full CFD runs, others up to 1000 at lower accuracy
- Applied and tested in a NPP relay room

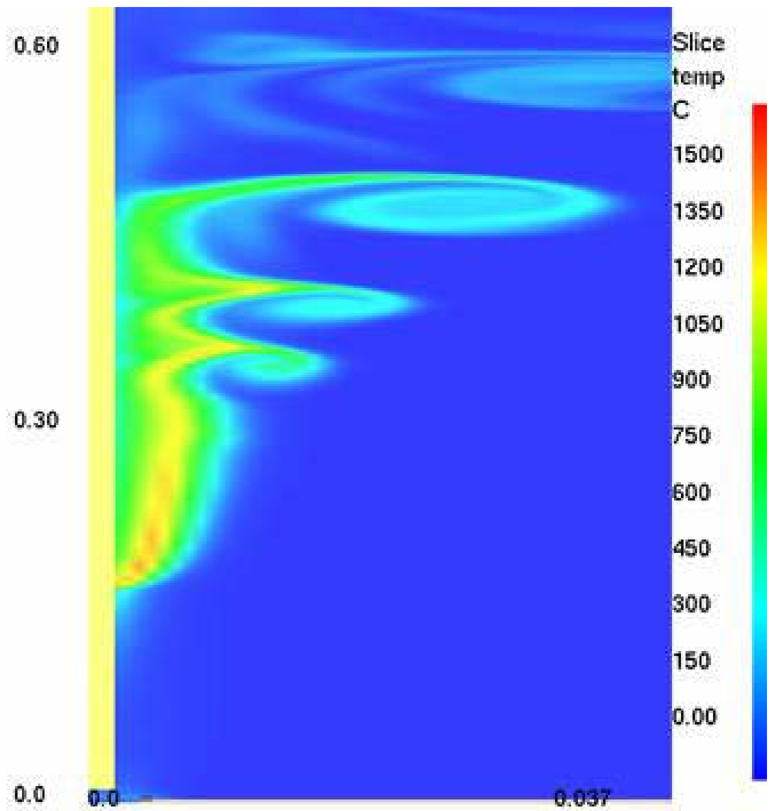


Contact person: Simo Hostikka, VTT, Simo.Hostikka@vtt.fi

Flame spread modelling

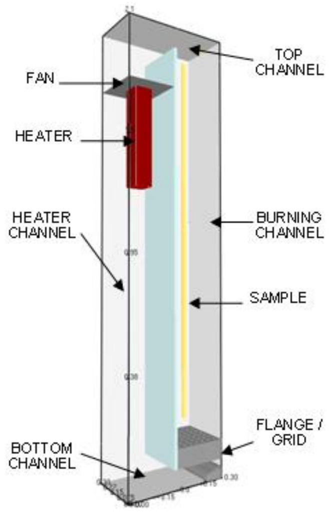
A new vertical flame spread model has been depicted

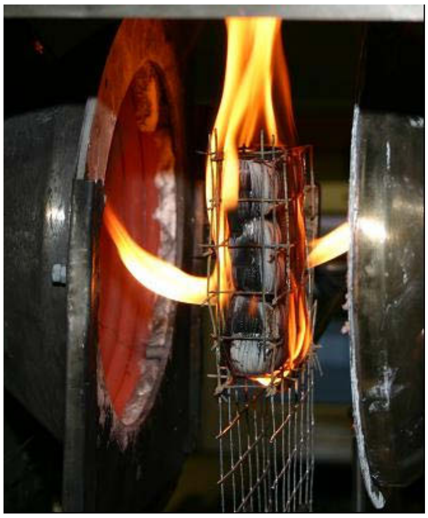
- Using literature studied
- By many quick, small scale experiments
- By direct numerical simulation (DNS) solving of Navier-Stokes equations
- Engineering flame spread subgrid model for FDS simulation code under construction using simplified heat transfer modeling



Contact person: Simo Hostikka, VTT, Simo.Hostikka@vtt.fi

New flame spread measuring instruments

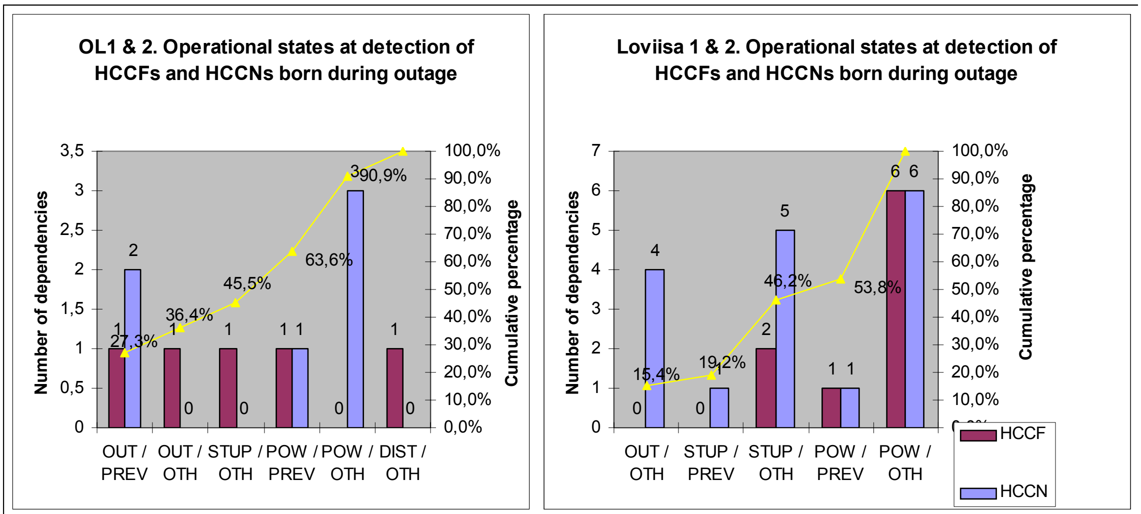




- A new 2 m vertical sample test rig for flame spread velocity as a function of initial temperature (left)
- Double cone calorimeters for energy and mass balance (right)

Contact person: Johan Mangs, VTT, johan.mangs@vtt.fi

Systematic analysis of human common cause failures in relation to maintenance activities



Type of detection: periodic testing or preventive maintenance (PREV), otherwise (OTH)
Operating states at detection: power operation (POW), refuelling outage prior to start-up (OUT), hot shutdown (HOT), cold shutdown (COLD), plant start-up (STUP), plant shutdown (SHDN)

- Identification and analysis of single errors and common cause failures from the maintenance history
- Recommendation for prevention of errors and their effects

Contact person: Ilkka Männistö, VTT, ilkka.mannisto@vtt.fi

Risk-informed ways of management of fire situations

RESEARCH TASKS

- Definition of the most important management tasks requiring co-operation in fire situations
- Identification of the influence of the ways of co-operating on the controllability of fire situations
- Identification of means to support risk-informed co-operation

How to achieve integrated support for collaborative safety management task?

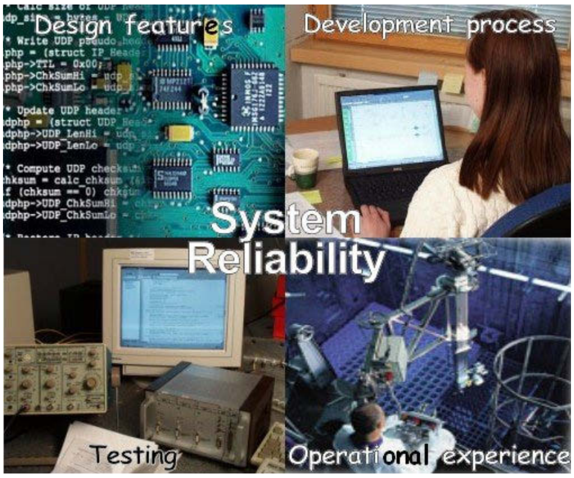
BENEFITS IN

- Development of a shared plant-specific concept of risk-informed management of fire situations
- Development of integrated operating procedures and training for the co-operational parties
- Fire risk analysis

Contact person: Kristiina Hukki, VTT, kristiina.hukki@vtt.fi

Reliability estimation of computer-based systems using Bayesian networks

- Development of a quantitative reliability estimation method of computer-based systems operating in safety-critical applications
- Bayesian networks are applied in the integration of different evidence including expert judgements and operating experience
- Two case studies on software reliability of a motor protection relay
- Literature review of statistical software reliability assessment methods



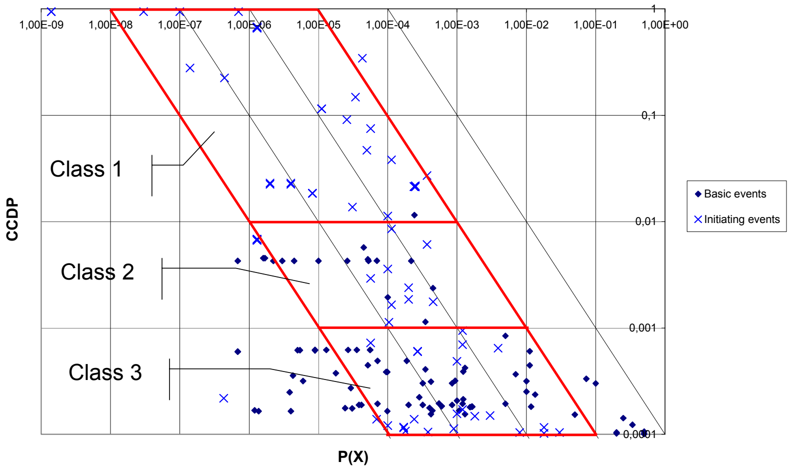
BENEFITS IN

- Quantitative reliability estimates for probabilistic safety assessment
- Licensing of systems for safety-critical applications

Contact person: Urho Pulkkinen, VTT, urho.pulkkinen@vtt.fi

Risk-informed categorisation of systems, structures and components

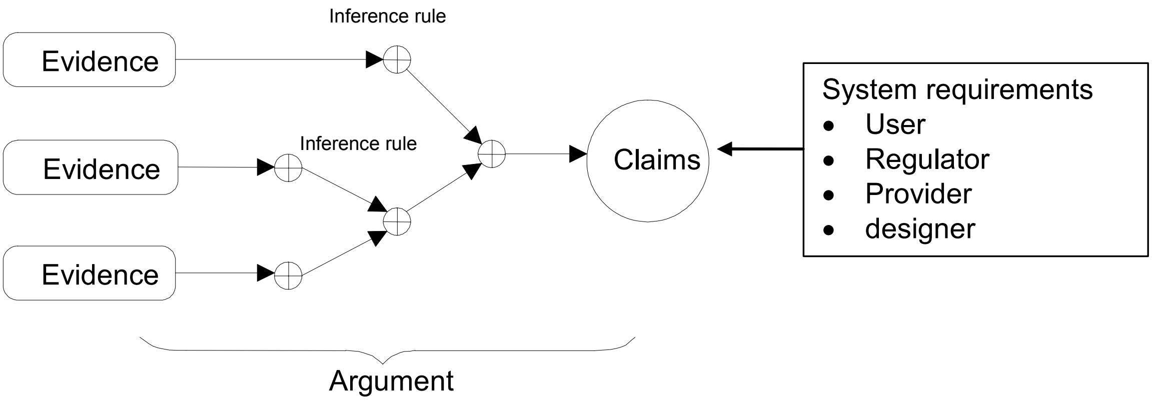
- Use of risk importance measures for risk-informed categorisation
- Targeting for safety improvements, i.e. using the probabilistic importance measures to decide between increased reliability or increased redundancy



Contact person: Ilkka Männistö, VTT, ilkka.mannisto@vtt.fi

Assessment of smart device software, ASDES

A goal based method for safety justification that defines claims, elaborates and apportions them to smart devices and components and then creatively identifies the arguments required to show these claims was developed.



- Analysis of Finnish regulatory requirements (mainly YVL 5.5 Guide)
- A real smart device and two application cases from Finnish nuclear power plants as a reference case study
- A generic safety case for smart devices was developed

Contact person: Urho Pulkkinen, VTT, urho.pulkkinen@vtt.fi

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Title

SAFIR. The Finnish Research Programme on Nuclear Power Plant Safety 2003–2006

Executive Summary

Abstract

Major part of Finnish public research on nuclear power plant safety during the years 2003–2006 has been carried out in the SAFIR programme. The programme has been administrated by the steering group that was nominated by the Ministry of Trade and Industry (KTM). The steering group of SAFIR has consisted of representatives from Radiation and Nuclear Safety Authority (STUK), Ministry of Trade and Industry (KTM), Technical Research Centre of Finland (VTT), Teollisuuden Voima Oy (TVO), Fortum Power and Heat Oy, Fortum Nuclear Services Oy (Fortum), Finnish Funding Agency for Technology and Innovation (Tekes), Helsinki University of Technology (TKK) and Lappeenranta University of Technology (LTY).

The key research areas of SAFIR have been 1) reactor fuel and core, 2) reactor circuit and structural safety, 3) containment and process safety functions, that was divided in 2005 into 3a) thermal hydraulics and 3b) severe accidents, 4) automation, control room and IT, 5) organisations and safety management and 6) risk-informed safety management. The research programme has included annually from 20 up to 24 research projects, whose volume has varied from a few person months to several person years. The total volume of the programme during the four year period 2003–2006 has been 19.7 million euros and 148 person years.

The research in the programme has been carried out primarily by Technical Research Centre of Finland (VTT). Other research units responsible for the projects include Lappeenranta University of Technology, Fortum Nuclear Services Oy, Helsinki University of Technology and RAMSE Consulting Oy. In addition, there have been a few minor subcontractors in some projects.

The programme management structure has consisted of the steering group, a reference group in each of the seven research areas and a number of ad hoc groups in the various research areas.

This report gives a short summary of the results of the SAFIR programme for the period January 2003 – November 2006, and highlights of some major achievements.

Keywords
nuclear safety, reactor components, reactor core, fuel elements, high-burnup, thermal hydraulics, severe accidents, containment, reactor circuit, structural safety, control rooms, automation, organisations, risk informed safety management, ageing, reactor analysis

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The deliverables of the programme include 545 publications, 6 Doctoral degrees, 1 Licentiate degree and 17 Master's degrees. The total volume of the programme during the four year period 2003–2006 has been 19.7 million euros and 148 person years.

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