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Nordic nuclear safety research

NKS-107
ISBN 87-7893-166-5

Final Report of the "Nordic Thermal-Hydraulic and Safety Network (NOTNET)" - Project

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April 2005

Abstract

A Nordic network for thermal-hydraulics and nuclear safety research was started. The idea of the network is to combine the resources of different research teams in order to carry out more ambitious and extensive research programs than would be possible for the individual teams. From the very beginning, the end users of the research results jave been integrated to the network. The aim of the network is to benefit the partners involved in nuclear energy in the Nordic countries (power companies, reactor vendors, safety regulators, and research units.

Key words

Thermal-hydraulics, nuclear safety, Nordic network, NOTNET

NKS-107
ISBN 87-7893-166-5

Pitney Bowes Management Services Denmark A/S, 2005

The report can be obtained from
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FINAL REPORT OF THE "NORDIC THERMAL-HYDRAULIC AND NUCLEAR SAFETY NETWORK (NOTNET)" –PROJECT

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Publicity: Public

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Diary code (VTT) PRO1-79T-03	**Order reference** AFT/NKS-R(04)35/9
Project title and reference code 12NOTNET	**Report identification & Pages** PRO1/P7013/04 29 p. + App. 7 p. **Date** 31.8.2004

Report title and author(s)

FINAL REPORT OF THE "NORDIC THERMAL-HYDRAULIC AND NUCLEAR SAFETY NETWORK (NOTNET)" -PROJECT

Tuunanen, J., Tuomainen, M.

Summary

A Nordic network for thermal-hydraulics and nuclear safety research was started. The idea of the network is to combine the resources of different research teams in order to carry out more ambitious and extensive research programs than would be possible for the individual teams. From the very beginning, the end users of the research results have been integrated to the network. Aim of the network is to benefit the partners involved in nuclear energy in the Nordic Countries (power companies, reactor vendors, safety regulators, research units).

First task within the project was to describe the resources (personnel, know-how, simulation tools, test facilities) of the various teams. Next step was to discuss with the end users about their research needs. Based on these steps, few most important research topics with defined goals were selected, and coarse road maps were prepared for reaching the goals. These road maps will be used as a starting point for planning the actual research projects in the future.

The organisation and workplan for the network were established. National coordinators were appointed, as well as contact persons in each participating organisation, whether research unit or end user. This organisation scheme is valid for the short-term operation of NOTNET when only Nordic organisations take part in the work. Later on, it is possible to enlarge the network e.g. within EC framework programme.

The network can now start preparing project proposals and searching funding for the first common research projects.

Distribution	**Publicity**
Petra Lundström (Fortum) 2 kpl	Public

Project manager	**Reviewed and approved by**	
Minna Tuomainen	Eija Karita Puska Group Manager	Seppo Vuori Research Manager

CONTENTS

1 INTRODUCTION

In Finland and Sweden, Universities (Chalmers, KTH, and LUT), Research Centres (VTT, Vattenfall Utveckling), nuclear safety regulators (SKI, STUK) and reactor vendors (Westinghouse Electric Sweden) are all carrying out research and development in the field of nuclear reactor thermal-hydraulics and safety. Traditionally, the national research programmes, such as FINNUS or SAFIR in Finland, or APRI in Sweden, have formed the forum for the national research projects. Common EC or NKS projects or commercial contracts have formed the way for co-operation between Nordic countries.

The first ideas of starting a Nordic thermal hydraulic and nuclear safety network were presented in the meetings between Westinghouse Atom and VTT. Based on these discussions, a joint meeting of Nordic TH-center & Condensation Pool Phenomena was organised at Espoo, Finland, Thursday, 21 August, 2003[1]. In this meeting, a short introduction of the research resources in different organisations was presented. In 2003, an application to NKS for a start-up project of Nordic Thermal-Hydraulic and Nuclear Safety Network (NOTNET) was sent and accepted for funding. As a part of this NKS project, the second meeting was organised in Västerås, Sweden in 11.3.2004[2]. The main purpose of the meeting was to discuss the possibilities to start a Nordic thermal-hydraulic and nuclear safety network, which would initiate and carry out common research projects for the benefits of all participating organisations.

The purpose of this report is to present a summary and outcome of the second meeting and to present the organisation and work plan of the network. The report also includes a description of available resources and equipment in the participating organisations for thermal-hydraulic and nuclear safety experiments and analyses.

2 NOTNET PROJECT DESCRIPTION

The objective of the NKS NOTNET project is to start a **Nordic Thermal-Hydraulic and Nuclear Safety Network (NOTNET)**, which discusses and develops common research projects in the area of thermal-hydraulics, severe accidents and nuclear safety in general. The objectives of the common projects is to make significant progress in the selected research areas for the benefit of all nuclear operators in the Nordic countries (*safety regulators, reactor vendors, research organisations and power companies*). An example of a potential thermal-hydraulic research area is the evaluation of dynamic loads to the pressure suppression pool structures during blowdown of steam or non-condensable gases. This issue is important because it affects greatly on the dimensioning of containment structures in BWRs. An example of potential severe accidents research area is the coolability of an ex-vessel corium melt pool, interacting with concrete. As a third example it can mentioned that significant progress is needed in understanding the multiphase flows and heat transfer in fuel assemblies of LWRs, including the critical heat flux (dryout and DNB) phenomena.

The objective of this project is to

- evaluate and report the infrastructure facilities, measurement techniques, modelling capabilities available in participating countries,
- discuss with the end users (utilities, authorities, Nordic Owners Group, …) of research priorities,
- propose an organisation and working procedures for the center,
- investigate the funding possibilities,
- develop the first work and action plan for the network, and
- start preparation of the first research projects within the network.

The output of the project is a roadmap type of research plan based on the end-user's needs.

The NOTNET project included five Tasks:

In the first task (**TASK 1**), information was collected about the existing infrastructure, such as experimental facilities, measurement techniques, modelling capabilities, calculation methods, man-power, available in different countries. This is discussed in the Chapter 3 of this report.

In the second task (**TASK 2**), it was discussed nationally about the interests of different end-users to participate in the network. At the same time, the research capabilities in different organisations were evaluated, and research priorities were discussed. This is discussed in the Chapter 4 of this report.

In the third task (**TASK 3**), a meeting was organised where the feedback from national organisations was discussed, and a preliminary work and action plan for the network was prepared. In the same meeting, the organisation of the network activity and the first potential common projects were discussed. As mentioned earlier, the meeting was organised in Västerås, Sweden. This is discussed in the Chapter 5 of this report.

The fourth task (**TASK 4**), the common research projects, was only started within the NKS project. The common projects were specified within the network after discussions with national end-users of the research results.

The final task of NOTNET project (**TASK 5**) included preparation of this final report of NOTNET project.

3 EXISTING INFRASTRUCTURE IN THERMAL HYDRAULICS AND NUCLEAR SAFETY

The existing infrastructure was reviewed in the first phase of the project in the Espoo meeting[1] and later in the NOTNET meeting in Västerås[2]. In the following chapters, a short overview of the existing infrastructure (experimental capabilities, theoretical analyses tools and manpower) in Chalmers, KTH, LUT, VTT, Westinghouse and Vattenfal Utveckling is presented. A general description of different organisations is presented in the appendices of this report.

3.1 KTH NUCLEAR REACTOR TECHNOLOGY

The equipment and experimental rigs include

- High-pressure (up to 25 MPa) two-phase flow loop for heat transfer, dryout and post-dryout studies in annuli and tubes.
- Plexiglas mockup of 5x5 rod bundle for air-water tests.
- MAGNE loop for natural circulation studies

The theoretical skills include

- CFD code CFX
- Thermal hydraulic system codes RELAP5, PARCS, TRACE.

The personnel at the laboratory includes

- 1 professor, 6 PhD Students, 2 associates/visitors, 1 technician, 1 secretary

3.2 KTH NUCLEAR POWR SAFETY

The equipment and experimental rigs include

- Several large and small scale test facilities for severe accident and advanced reactor studies
- Two concrete reinforced pressure containments with up to static pressure of 6 bars, 4x4x4 m^3 in volume and 400 and 600 mm thick.
- Two high-speed photography systems with maximum frame speeds of 8000 and 100,000 fps.
- High-speed continuous x-ray radiography system with two high-speed cameras for advanced measurement of volume fractions of components in the multi-component multi-phase flow.

The theoretical skills include

- In-house CFD tools
- Free surface moving boundary problems: SIPHRA-3D
- Lattice-Boltzmann CFD method: FLOWLAB
- Melt pool convection and quenching problem: MVITA, COREQUENCH
- Integrated melt-coolant interaction: COMETA
- Thermal hydraulic system codes, severe accident and reactor core analyses tools: RELAP5, SAS4A, MELCOR, COBRA-IV, TRACE, PARCS.

The personnel at the laboratory includes

- 1 Senior permanent (prof. + lecturers), 4 Post-doc researches, 5 PhD students, 3 MSc students, 2 associates/visitors, 3 technicians, 1 secretary

3.3 LAPPEENRANTA UNIVERSITY OF TECHNOLOGY (LUT)

The experimental research at LUT started in 1970's with reflooding tests on REWET-I and REWET-II test rigs. The work was first carried out as a part of VTT but in 2002, the experimental team was moved to LUT. In 2004, when Fortum is giving up experimental work at it's hydraulic laboratory, there has been discussions of transporting some of the experimental tools from Fortum's laboratory to LUT.

The experimental capabilities in the current laboratory include

- PACTEL test rig for VVER and PWR (if SGs are turned to vertical position) thermal hydraulic studies including SBLOCA and transients,
- Different pressure vessels and pools, used in several test programmes (SWR1000 hydraulic scram and boron injection system tests),
- VEERA test rig for boron mixing and precipation studies, used also for reflooding studies,
- Condensation pool test rig (gas or steam tests with or without pump and strainers),
- BWR 90+ isolation condenser and core catcher test rigs.

The theoretical skills include

- Possibility to use CFD codes (FLUENT or others),
- Possibility to use international or national thermal hydraulic codes (RELAP5, CATHARE, APROS, NORCOOL, ...),

The personnel at the nuclear safety research centre includes

- 4 researchers, 1 engineer, 3 technicians and 1 MSc student.

The group works together with the professor of Nuclear Technology at LUT, who has a group of

- 1 professor, 1 laboratory engineer, 2 assistants.

3.4 CHALMERS

The research group is located in a department within the School of Physics and it is a part of Chalmers Centre for Nuclear Technology (CKTC).

The experimental capabilities include

- Pulsed neutron generators and a pulsed beam for slow positrons.

The analytical tools include

- Several reactor physics codes: CASMO-4, TABLES-3, SIMULATE-3, PARCS, MCNP-4C.
- Thermal-hydraulic system codes: RELAP5 with PARCS, TRACE.
- CFD codes: FEMLAB-3, FLUENT.

The personnel at the laboratory includes

- 4 permanent academic (prof+lecturers), 1 Post-doc researcher, 4 PhD students, 2 MSc students, 1 visiting scientist, 2 technicians and 1 secretary.

3.5 VTT

The nuclear research in VTT is carried out in different research institutes, but the research is co-ordinated under the VTT Nuclear portal. In the thermal-hydraulic area, VTT is concentrating on using and developing tools for nuclear reactor safety analysis.

The experimental capabilities in severe accidents include

- Particle bed dryout test rig,
- A test rig for aeroplane crash studies (under construction as a part of SAFIR programme).

The theoretical skills include

- several reactor physics codes, both own development and international (CASMO, SIMULATE, CROCO, ARES, ENIGMA, FRAPCON, TRAB-3D, HEXTRAN, GENFLOW, FRATRAN)

- Several CFD codes (FLUENT, CFX, STAR CD), with a possibility to connect the codes to structural analysis tools (ABAQUS)
- International thermal-hydraulic and severe accident system codes (RELAP5, CATHARE, MELCOR, SCDAP/RELAP, CONTAIN, DET3D)
- Own thermal-hydraulic, structural analyses and severe accident codes (APROS, PASULA, PIPING).
- Codes with coupled Thermal-hydraulics and 3D neutronics (APROS, TRAB-3D)

The personnel at the laboratory includes

- VTT Nuclear has about 260 researchers in different VTTs units.

3.6 VATTENFALL UTVECKLING

Vattenfall Utveckling has operated in Elvkarleby since 1943 and the laboratories serve the whole Vattenfall group.

The experimental capabilities include

- Vattenfall Utveckling AB has own fluid mechanics lab.

The theoretical skills include

- Vattenfall Utveckling AB has experience in using various commercial CFD codes, e.g. Fluent and Star-CD.

The personnel at the laboratory includes

- 150 employees working in various fields, including the fluid mechanics and heat transfer field.

4 RESEARCH PRIORITIES OF END USERS

The research priorities were evaluated in the NOTNET meeting in Västerås. Some comments were also received before and after the meeting from Fortum and TVO, Finland.

4.1 FORSMARK KRAFTGRUPP AB (FKA)

The research priorities of FKA were presented in the Västerås meeting[2]. From FKA's point of view, the most important issues are connected to core area and the main activity of NOTNET should be in these topics. In the actual research projects, the people from nuclear power plants should be involved in the work.

The high priority research tasks and development needs included

- Improved modeling and prediction of dryout margins, including post-dryout capabilities
- Modeling of cross-flows in reactor cores
- Prediction of flow distribution between various assemblies in reactor cores
- Modeling and assessment of various flow instability modes
- Flow stratification and concentration transport and distributions
- Distribution of loads on structures coming from different sources: blow-down to pool, etc
- Severe accidents: understanding of steam explosions; cooling of reactor to keep corium inside the vessel
- PCI – improved methods to evaluate the risk of occurrence
- Effects of control rods and boron

4.2 OKG AKTIEBOLAG (OKG)

The research priorities of OKG were presented in the Västerås meeting[2]. The research tasks and development needs included

- Understanding and modeling of fluid flow induced vibrations; non-stationary, coherent structure dynamics
- Evaluation of dryout risk
- Prediction of flow distribution in a core
- Prediction of natural circulation patterns in a reactor pressure vessel

- Criteria for detection and suppression of power oscillations
- Condensation pool performance
- Modeling of transient two-phase flows

4.3 STATENS KÄRNKRAFTINSPEKTION (SKI)

The research priorities of SKI were presented in the Västerås meeting[2]. SKI can participate projects, which aim at developing new knowledge and skills. In general, SKI's funding for research is not increasing. The following topics within nuclear reactor thermal-hydraulics and safety have the highest priority for SKI:

- Support national competence in the field of reactor safety
- Support nuclear inspections
- Methods to evaluate power uprates and modernizations
- Coupling between TH modules with neutronic modules
- Accuracy of 1D and 3D predictions
- Power plant aging

4.4 WESTINGHOUSE ELECTRIC SWEDEN

The research priorities and skills of WES were presented in the Västerås meeting[2]. The high priority research tasks and development needs include

- Modeling of two-phase flow in fuel assemblies
- Mechanistic modeling of dryout and DNB; improved evaluation of thermal margins
- Modeling of post-dryout
- Modeling of transient (oscillatory) two-phase flows with boiling transition
- Condensation pool dynamics
- Coupling of nodal codes with CFD codes
- Development of BE codes for ATWS analysis

4.5 FORTUM POWER AND HEAT

The research priorities of Fortum Power and Heat were received through an E-Mail before the Västerås meeting[3]:

- Prediction of stagnation of single-phase natural circulation in VVER reactors during low-power operation (connection to pressurized thermal shock, PTS)

- Connection of APROS thermal-hydraulic code with CFD tools
- Connection of CFD and structural analysis tools (see for example SAFIR Multiphysics -project)
- Dynamic loads to reactor pressure vessel internals (for example LBLOCA)
- water hammer phenomena and analysis
- Thermal hydraulic issues relevant for plant life management
- Flow induced vibrations
- Stratification phenomena and prediction of stratification with analytical tools, such as CFD

4.6 TEOLLISUUDEN VOIMA OY (TVO)

The research priorities of TVO were received through an E-Mail after the Västerås meeting[4]:

- Suppression pool behaviour in BWRs
- Phenomena in LBLOCA of PWRs
- Behaviour and modelling of an open core of a large PWR, focusing on the cross flows inside the core

5 ORGANISATION AND WORKPLAN FOR NOTNET

5.1 BACKGROUND

Since 1989, research on nuclear safety, waste management and structural safety in Finland has been organised in national research programmes (YKÄ, RATU, RETU, FINNUS, SAFIR, JYT, KYT, etc.). In these programmes, VTT and Lappeenranta University of Technology have been the main research organisations in the thermal-hydraulics area. The role of the industry and regulator has been very active including specification of the research projects and objectives, participation in the projects and financing of the work.

The current SAFIR programme is designed for the years 2003-2006 and it includes projects with different funding sources (VYR, Tekes, VTT, Fortum, TVO, …). The steering group of SAFIR evaluates the applications annually but the projects may be longer. In Finland, the public research programmes offers a natural basis for the NOTNET co-operation. In fact, some Swedish organisations are already taking part in the SAFIR projects.

In Sweden, mainly KTH, Chalmers and Westinghouse (only internal) perform research on nuclear thermal-hydraulics and safety. EU programs, SKI and SKC provide funding for the work. Further coordination of projects is done between different divisions at KTH (through CEKERT) and at Chalmers. Proposals are submitted on a regular basis and evaluated by SKC. Projects can take several years (up to 4 years as a PhD-thesis).

5.2 BENEFITS OF NOTNET

In the small European countries that use nuclear energy to produce electricity, such as Finland and Sweden, there are many excellent research teams and equipment, but the teams are typically small and could not cover all, important aspects of nuclear safety. For example, KTH in Sweden has a strong team on experimental and analytical research of severe accidents. KTH has co-ordinated several EC research projects and performed a large number of unique tests. In the same way, LUT in Finland has a long experience on thermal-hydraulic system and component tests for VVERs and ALWRs. Recently, LUT performed several tests in co-operation with the vendors, which were offering new reactors for TVO.

Gaps in the research may also exist in larger European countries, such as Germany, where political reasons have reduced research funding and interest of young engineers for nuclear technology. NOTNET can bring together the Nordic researchers, to form larger research groups and to fill the gaps in the research capabilities in the individual countries. In the longer term, the intention is to widen the co-operation to the countries in central Europe, especially to those countries having BWRs in operation.

5.3 ORGANISATION OF THE NETWORK

This chapter presents a possible organisation of NOTNET activities. This organisation scheme is valid for the short-term operation of NOTNET when only Nordic organisations take part in the work.

5.3.1 Organisation in Finland

The basic idea of the organisation of NOTNET work is to organise it within the existing structures, such as SAFIR programme in Finland. This has an advantage that it would not create any new, additional bureaucracy. All Finnish organisations operating within the nuclear safety are already involved in SAFIR. SAFIR has a working organisation for evaluation of the project proposals. SAFIR research seminars, which are held every two years, offer a forum for presenting the results of NOTNET projects.

5.3.2 Organisation in Sweden

The principles of the network organisation in Sweden are shown in the figure below.

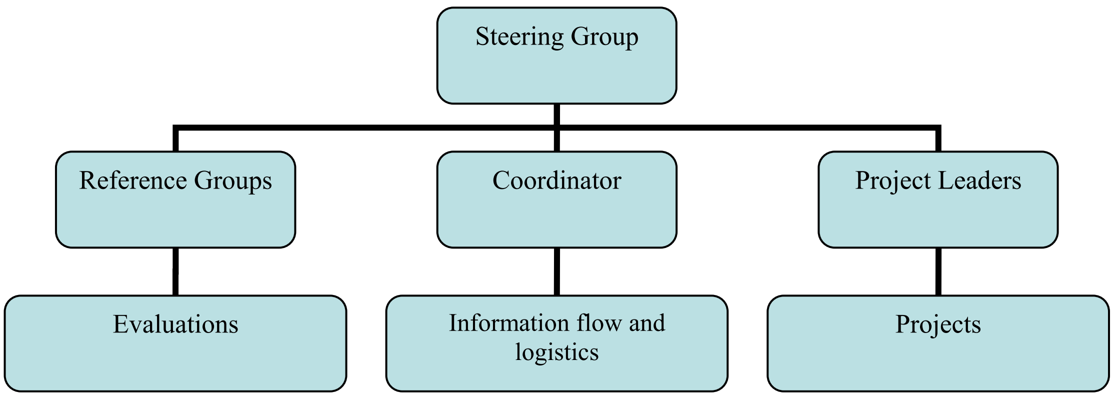


Figure 1: Network organisation in Sweden

The Steering Group will be formed by End Users (typically each End User will be represented by a single person in the Steering Group). Its role will be to prioritize various research directions performed within NOTNET as well as to make decisions concerning financial support for chosen projects. It will also evaluate the final result of projects based on the opinions prepared by Reference Groups. The Steering Group will have regular meetings on 2-to-3-times-a-year basis.

Reference Groups will evaluate project proposals and projects. Each project proposal and each project will have its own reference group. The primary goal of reference groups will be to provide proper information to the steering group concerning the required research direction as well as progress in individual projects. The members of the reference groups should be experts in the field and should preferably be designated by end users. One person can be a member of several reference groups. Reference groups will meet only when needed, typically to evaluate a project proposal or to evaluate a project status.

Project Leaders will be responsible for conducting projects approved by the Steering Group. Project Leaders will usually be persons originating from the

5.3.3 Coordination

For coordination of NOTNET work, national coordinators are nominated. The coordinators will be responsible for contacts and exchange of information between countries within NOTNET (in the beginning Finland and Sweden), between NOTNET and domestic related organisations and later on maybe between NOTNET and EU.

Since the research should be based on the real need of the end-users of the results, contact persons in the end user organisations are also nominated. The present contact persons are listed in Appendix 5.

During the NOTNET operation, it’s important to organise the funding of the co-ordination work. The amount of co-ordination work depends on the amount of activities under way, but it can be estimated to be about 1-2 man-months of work per year in the participating countries. The participants and/or the public sources should finance this work.

5.4 WORKPLAN FOR NOTNET

A possible way to organise the NOTNET work is presented in *Figure 2*.

The operation of the network begins with the specification of **research needs and priorities**. In this phase, the end-users of the research results, such as power companies and regulators, should be involved in the work. The research needs and priorities can be discussed and finalised in NOTNET meetings and workshops. Public sponsors, such as NKS or EC, for such workshops are already available. Based on these discussions in the workshops, some research topics should be selected for high priority topics.

Based on the research priorities and needs, **roadmaps to describe how the research needs are fulfilled should be done**. The national co-ordinators in Finland and Sweden can make the preliminary roadmaps, and send them for comments to the end-users. The outcome of the work is the final roadmaps for the selected research topics.

The third step of the work is **the preparation and evaluation of the project proposals**. The proposals should be such that they are closely connected to the roadmaps. Each realised project should make a step towards the final goal of the roadmap. In this way, it would be possible to guarantee long term funding of the work. National co-ordinators can submit the proposals for national funding organisations. In some cases, it's also possible to apply funding from the EC.

The fourth step is the **actual research projects**. The outcome of the projects is the research reports, new analysis methods or new experimental capabilities. The results of the projects can be presented in NOTNET meetings and workshops.

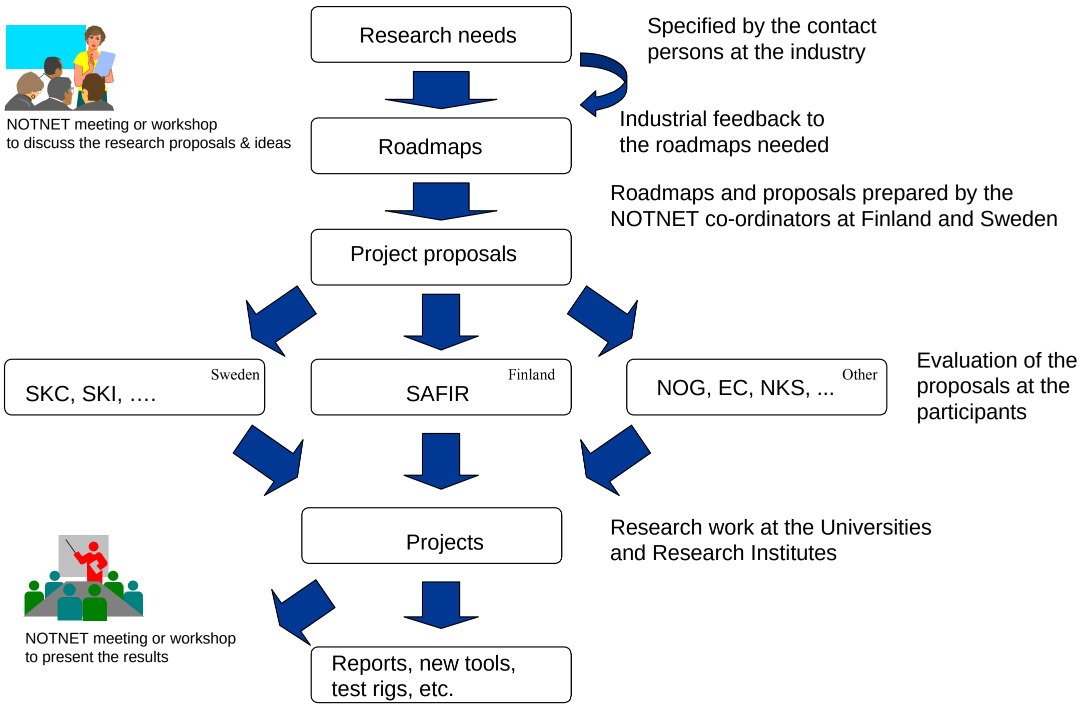


Figure 2: General view of organisation of NOTNET work.

5.5 FUNDING OF COOPERATION

The research projects may later, outside the proposed NKS project, search funding from national or international sources, such as

- EC 6th framework programme,
- Finnish national research programmes, such as the current SAFIR programme,
- NKS-R research programme,
- Nordic owners group NOG,
- Technology Development Centre (TEKES) in Finland,
- APRI project

6 ROADMAPS FOR FUTURE RESEARCH PROJECTS

In the following chapters, a short description of roadmaps for three research topics, driven by the end user needs, is presented. The proposal is based on discussions in the Västerås meeting[2] and it has been prepared in co-operation between VTT, KTH and Westinghouse[5]. The three maps below are based on the work already started or the work that is planned. Most of the research organisations in Sweden and Finland are involved in the work. The description includes also a preliminary co-ordinating organisation of the roadmap area and a list of possible participants in the work. The topics of the three road maps are

- Mechanistic modeling of two-phase flow and heat transfer in fuel assemblies
- Dynamic Condensation, Vessel Depressurization and Core Melt Loadings in BWRs
- Mixing and stratification phenomena in reactor pressure vessels

6.1 ROADMAP 1: FUEL ASSEMBLY FLOW AND HEAT TRANSFER

The project will be focused on the development of a computer code which will be capable to predict two-phase flow and heat transfer in fuel assemblies. The code will satisfy the following general requirements:

- Fully three-dimensional, both steady-state and transient, treatment of flow and heat transfer in fuel assemblies
- Capability to model all geometry details including spacers
- Capability to predict all single- and two-phase flow regimes encountered in fuel assemblies during normal and abnormal operation
- Capability to predict wall temperatures, void fraction distribution and pressure drop in fuel assemblies
- Capability to predict the onset of DNB and/or dryout and post-CHF heat transfer
- Modular programming that promotes easy implementation of models into existing commercial CFD codes

By “capability to predict..” is meant that the specific sub-model is based on mechanistic principles and is validated against proper and detailed measurements.

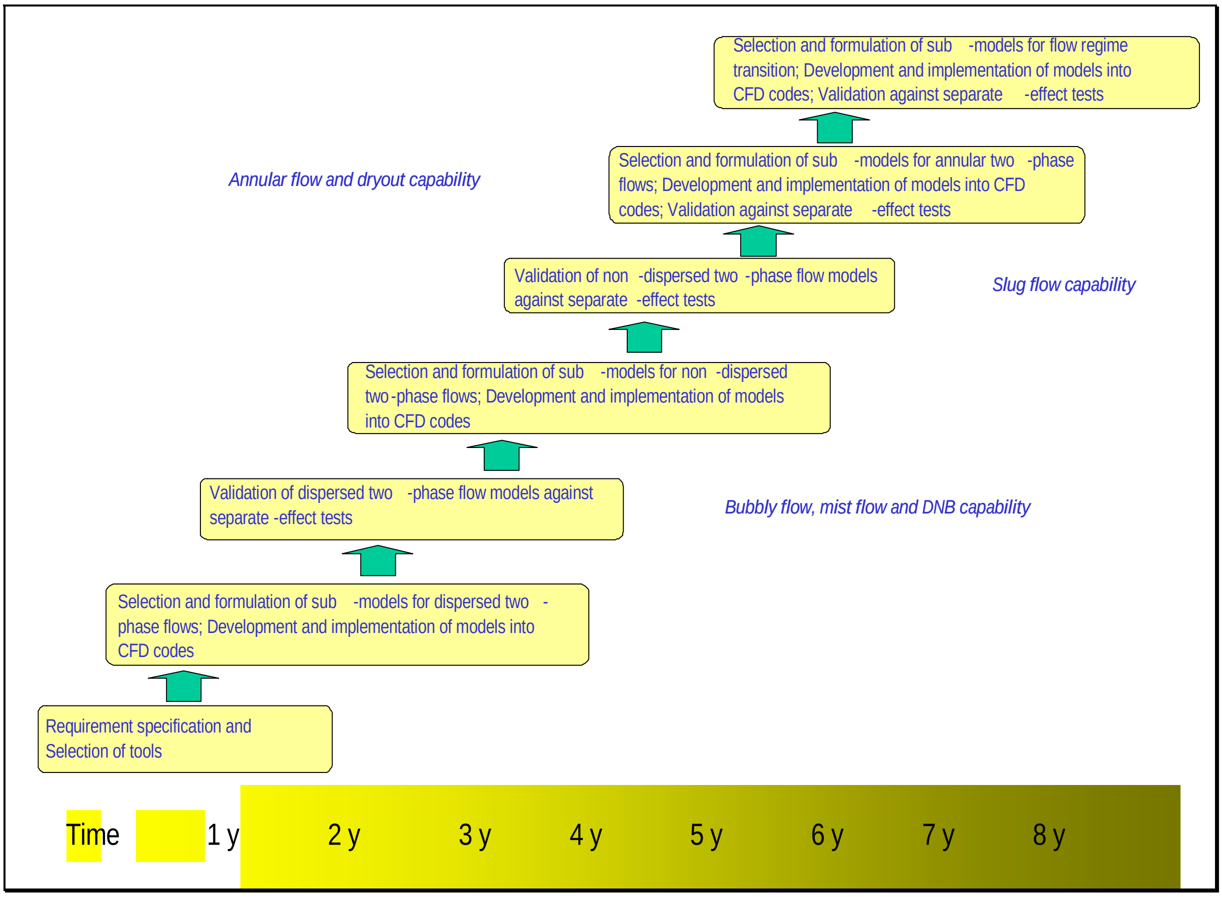


Figure 3: Roadmap 1, development of mechanistic model for two-phase flow in fuel assembles.

Table 1: Summary of objectives, benefits and existing skills in the field of mechanistic modeling of thermal-hydraulics and heat transfer in fuel assemblies

Lead:	KTH Reactor Technology
Participants:	KTH Reactor Technology, VTT, Westinghouse
Objectives:	– To develop and validate a detailed mechanistic modelling of two-phase flow and heat transfer in fuel assemblies. – To develop a new tool capable for 3-D analysis for all flow regimes starting from subcooled single-phase liquid flow and including subcooled and saturated boiling, film flow on fuel rods, dryout and post-dryout heat transfer. – To develop a new 3-D tool capable to calculate steady-state and transients including instability events.
Benefits:	– Increases competitiveness in the field – Promotes nuclear safety – Develops state-of-the-art CFD skills for design and safety analysis of fuel rod bundles – Develops advanced experimental technology skills – Enhances fuel design and safety assessments
Existing know-how and skills:	– Experimental and analytical work done at Westinghouse. Data from FRIGG loop available for validation (released with restrictions) – HEM development for Star-CD and Fluent under way in SAFIR INTELI and THEA projects (VTT) – Single-phase CFD analyses for VVER rod bundle *(P. Rautaheimo, E. Salminen, T. Siikonen and J. Hyvärinen, Turbulent Mixing between the VVER-440 Fuel Bundle Subchannels: A CFD Study, Proceedings of the Ninth International Topical Meeting on Nuclear Reactor Thermal Hydraulics, San Francisco, October 1999)* – Two-fluid CFD models for dispersed flow – Simple sub-cooled boiling models – Advanced Bundle Code (ABC) developed by Westinghouse for prediction of two-phase flow and heat transfer in fuel assemblies (*G. Windecker and H. Anglart, "Phase distribution in a BWR fuel assembly and evaluation of a multidimensional multifield model," Nuclear Technology, vol. 134, pp. 49-61, 2001*) – Spacer effects in single and two-phase flows (*H. Anglart et al. "CFD prediction of flow and phase distribution in fuel assemblies with spacers," Nuclear Engineering and Design, vol. 163, pp. 81-98, 1997*)
Research topics:	– Formulation and solution of governing equations with proper closure laws for multi-fluid flow and heat transfer – Implementation of two-phase material properties on selected CFD tools – Development of heat transfer, boiling and condensation models for selected CFD codes for different flow regimes – Separate effect experiments for developmental support and validation of the models

6.2 ROADMAP 2: DYNAMIC CONDENSATION, VESSEL DEPRESSURIZATION AND CORE MELT LOADS IN BWRS

During the last decades of research in the Nordic countries on the nuclear safety, rich resources such as experimental facilities and database were produced and accumulated. These infrastructures in the Nordic countries could be employed to pursue further research needed for resolving remaining key safety issues in terms of dynamic loadings to BWR vessel and containments during accidents. Three different dynamics loadings which may threaten the integrity of BWR vessel and containment are identified;

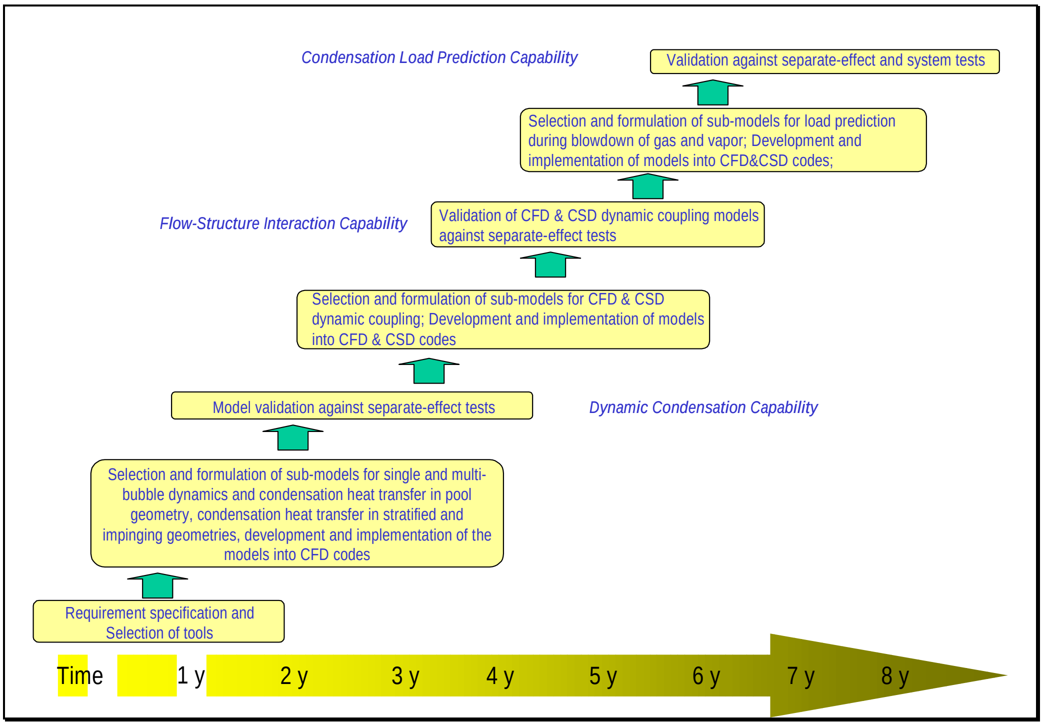
- Dynamic Condensation Loads in BWRs
- BWR Vessel Depressurization Loads
- BWR Core Melt Loads in Severe Accidents

6.2.1 Dynamic Condensation Loads in BWRs

Condensation inside channel or pool geometry is commonly encountered in many heat transfer devices and is of fundamental importance particularly in Light Water Reactors (LWRs). In nuclear reactors, direct contact condensation (DCC) of steam in subcooled liquid is a common event encountered in two-phase systems. In particular, DCC phenomena can significantly affect the course of a postulated Loss Of Coolant Accidents (LOCA) in LWRs. When steam comes into direct contact with subcooled water in several configurations, a steam-subcooled water interface plays a dominant role in determining heat transfer rates rather than cold wall. In BWRs, this condensation phenomenon occurs in pressure suppression system when steam is directly introduced to the subcooled condensation pool. This condensation process causes pressure and fluid oscillations and becomes one of important safety issues. The condensation mode is quite specific to the geometry and the mode of contact between steam and water besides the parameters of water subcooling etc.; stratified co-current and counter-current DCC in pipe geometry, and impinging/submerged DCC in pool geometry. The DCC phenomena specific to the Nordic BWRs will be modelled, in particular the configuration of the steam pipe and its submergence in the water.

The primary need in the field of the direct-contact condensation is to predict the condensation loads resulted from the dynamics of vapor bubbles in subcooled water in the pool geometry and the dynamic development of condensation interface and stability in a stratified contact mode. To that end it is necessary to develop proper models for bubble dynamics, condensation interfacial phenomena, as well as models to predict fluid-structure interactions with accurate quantitative measurement.

The present research will focus on the development of proper models which enable to predict the multiphase dynamics and heat transfer in various DCC contact modes, e.g., dynamics of steam or gas bubble in liquid pool, evolution and stability of condensation interface and fluid-structure interactions. This work inevitably requires phenomena-oriented experiments with appropriate scaling of the phenomena and accurate measurement techniques of key parameters in DCC.



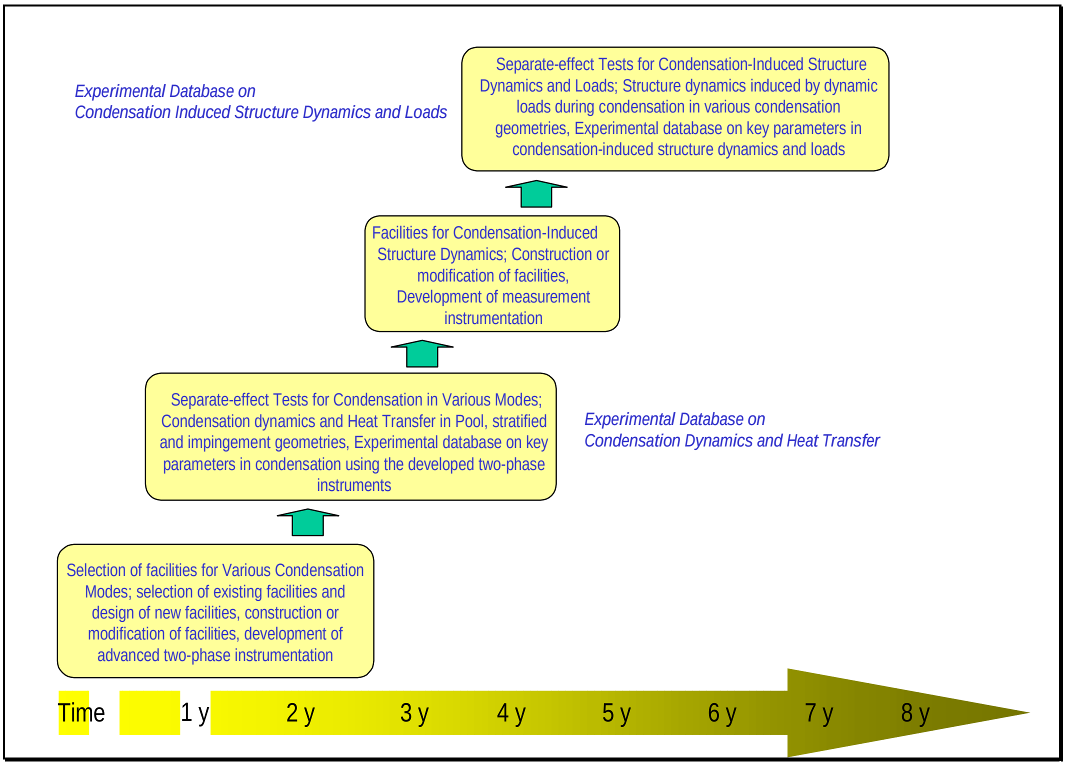


Figure 4: Roadmap 2a, Dynamic Condensation Loads in BWRs

6.2.2 BWR Vessel Depressurization Loadings

The safety studies of nuclear reactors are based on the analysis of the consequences of several hypothetical accidents. A large break Loss Of Coolant Accident (LOCA) caused by a major rupture within the primary cooling system is one of the accidents taken into consideration. Since the cooling system cools the nuclear reactor core, the assessment of this accident is essential for the reactor safety. When the primary cooling pipe is ruptured, the primary system coolant water is expelled through the break into the containment, causing the system pressure to drop in a few tens of milliseconds down to the highest local fluid saturation pressure. This violent depressurization process causes depressurization wave propagation through the primary cooling system and the reactor pressure vessel where it causes a dynamic loading of the internal core structures. In a small break LOCA in the primary cooling system, on the other hand, the rate of coolant escaping through a break is insignificant to cause a substantial deformation of the internal core structures. However, accurate knowledge of the coolant loss rate ("critical flow") through the break is important to predict the time limit until the core will be uncovered. In addition, for both small and large break LOCAs, the transient boiling from initial natural convection or partial nucleate boiling to film boiling on the heated core fuel rod caused by depressurization will be of importance.

No experimental data have been provided for the depressurization transients. In particular for Swedish BWR nuclear power plants, it is important to produce experimental database for this phenomena to assure the integrity of internal core and vessel structures and transient behaviors due to depressurization during small and large LOCAs. As for the Swedish BWRs, abnormal reactivity increase will be caused by the positive void feedback when transient boiling process from partial nucleate boiling to film boiling may increase the amount of steam at the reactor core. Hence, transient thermal-hydraulic characteristics on the core fuel assembly during the depressurization should be well investigated.

The work will focus on the investigation of depressurization transient and integrity of vessel internal structures due to depressurization during small and large break LOCAs.

- Scaling analysis.
- Depressurization Transient During Large Break LOCA.
- Depressurization Transient During Small Break LOCA.
- Critical Flow Analysis during the Depressurization.

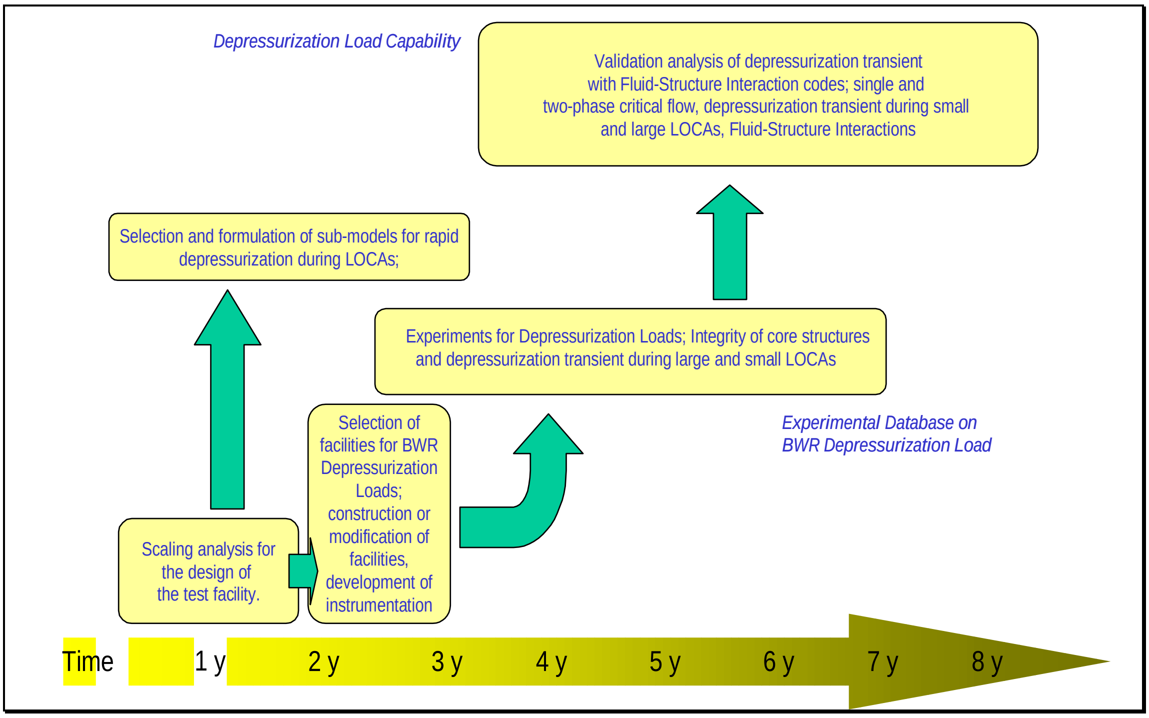


Figure 5: Roadmap 2b, BWR Vessel Depressurization Loads

6.2.3 BWR Core Melt Loadings in Severe Accidents

Reactor safety's concern with severe accidents, since the TMI-2 accident led to almost twenty years of intense research efforts, which have resolved a number of severe accident issues. Lately, research has been concentrated on accident management and a number of LWR plants, around the world, have adopted severe accident guidelines (SAMGs) and strategies. The challenging issues remained in reactor safety related to the design-based and severe accidents is to evaluate the dynamic and static loads imposed to the reactor vessel and containments during the progress of accidents. The several issues concerning core melt loadings in severe accidents have been assessed in the PRE-MELT-DEL project sponsored by NKS. This assessment addressed the remaining key issues in the order of priority which are of most interest to Nordic power companies and government regulatory organizations;

- In-vessel coolability of the melt pool or particulate debris.
- Ex-vessel coolability of the melt pool or particulate debris
- Energetics and fragmented debris characteristics of a steam explosions
- Characteristics of vessel failure.

The present work will focus on the experimental investigation of in- and ex-vessel coolability of the melt pool or particulate debris, steam explosions and vessel failure. Appropriate scaling methodology and theoretical models to predict in- and ex-vessel coolability, steam explosions in nuclear reactors during severe accidents will be developed based on the accumulated experimental database.

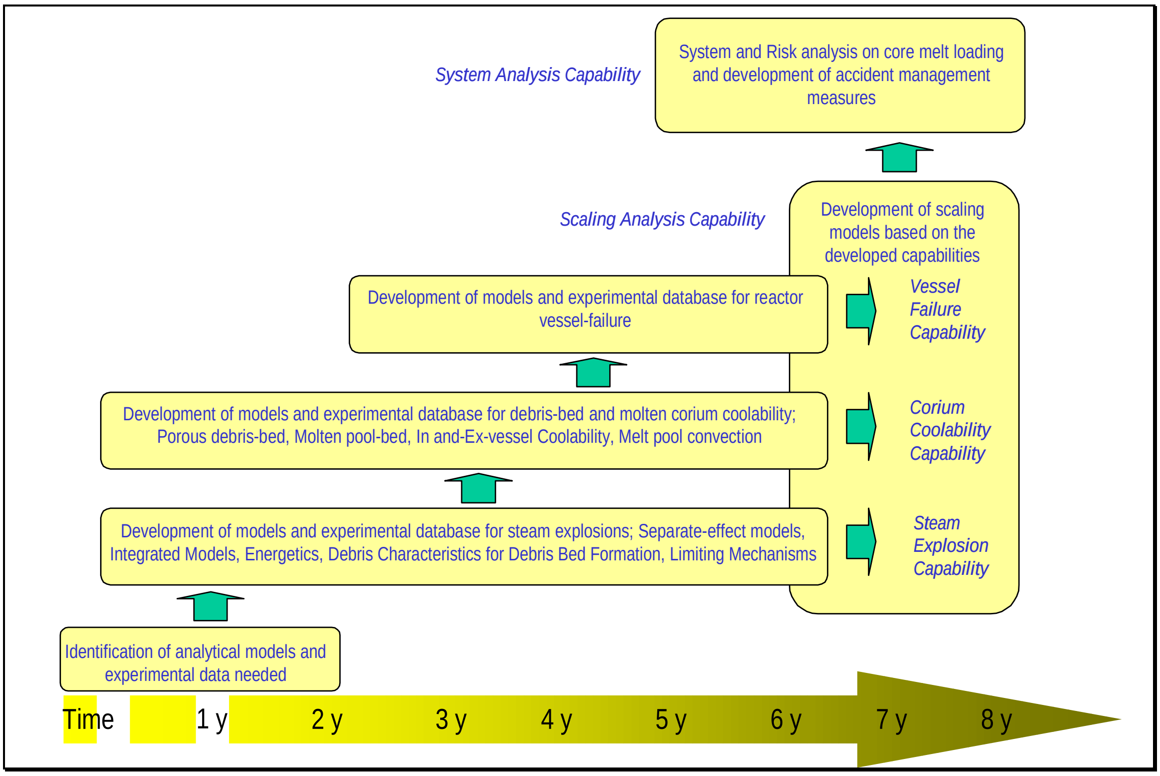


Figure 6: Roadmap 2c, BWR Core Melt Loads in Severe Accidents

Table 2 summarizes the objectives, benefits and existing skills in the fields of nuclear reactor safety concerning of static and dynamic loads originated by dynamic condensation in various geometrical modes, rapid depressurization during small and large LOCAs, and core melt formed in In and Ex-vessel during severe accidents.

Table 2: Summary of objectives, benefits and existing skills in the field of condensation pool dynamics and severe accidents

Lead:	KTH Nuclear Power Safety (Depressurization loads, Core Melt Loads) VTT (Condensation Loads)
Participants:	KTH Nuclear Power Safety, VTT, Westinghouse
Objectives:	<u>Dynamic Condensation Loads:</u> – To develop multidimensional simulation tools for the simulation of condensation phenomena in various condensation modes (submerged, impinging and stratified condensation) – To connect the CFD tools to structural analyses tools for analysing the forces on structures during dynamic condensation

	Vessel Depressurization Loads: – To develop the database to analyse the integrity of BWR internal systems and external loads during the depressurization transient. Core Melt Loads: – To develop multidimensional simulation tools for analysis of molten corium behaviour in the vessel and containment – To reduce uncertainties and risks connected to the severe accident issues of BWRs (*steam explosions, debris cooling in the containment and vessel and vessel integrityl*)
Benefits:	– Reduction of uncertainties and risk in the containment design – Enhancement of reactor safety – Advancement of the analytical and experimental techniques
Research topics:	Dynamic Condensation Loads: – Direct Contact Condensation Tests in Co-Current and Counter-Current Stratified Geometry (KTH/NPS) – Direct Contact Condensation Tests in Steam Impingement on Subcooled Water Pool (KTH/NPS) – Direct Contact Condensation Tests in Suppression Pool Geometry (LUT, KTH/NPS) – Development of condensation models for CFD codes (KTH/NPS) – Separate effect experiments for developmental support and validation of the CFD models (LUT, KTH/NPS) Vessel Depressurization Loads: – Direct Contact Condensation Tests in Co-Current and Counter-Current Stratified Geometry (KTH/NPS) – Direct Contact Condensation Tests in Steam Impingement on Subcooled Water Pool (KTH/NPS) – Direct Contact Condensation Tests in Suppression Pool Geometry (LUT, KTH/NPS) – Development of condensation models for CFD codes (KTH/NPS) – Separate effect experiments for developmental support and validation of the CFD models (LUT, KTH/NPS) Core Melt Loads: – In-Vessel Coolability Tests in Molten Pool and Particulate Debris (KTH/NPS) – Ex-vessel Coolability Tests in Molten Pool and Particulate Debris (KTH/NPS) – Steam Explosion Tests (KTH/NPS) – Vessel Failure Tests (KTH/NPS) – Particle bed dryout tests under plant specific parameters (VTT)
Existing know-how, skills and infrastructures:	Dynamic Condensation Loads: – Work performed to date by Westinghouse personnel – Methods and experiences available – Available data for validation – Analytical modeling on bubble dynamics with heat transfer (KTH/NPS) – Advanced free-surface modelling in two-phase flow (KTH/NPS) – Advanced measurement techniques (Void Probe, X-ray radiography) for condensation interface measurement (KTH/NPS)

	Vessel Depressurization Loads: – Similar work performed in the FOREVER program: Pressurized Vessel Failure Tests (KTH/NPS) – Fluid-Structure Analysis in the FOREVER program (KTH/NPS) Core Melt Loads: – Particle bed dryout tests as a part of SAFIR/SANCY project (VTT) – In- and Ex-vessel Coolability: POMECO, COMECO, DECOBI and SIMECO projects (KTH/NPS) – Steam Explosions: MISTEE program (KTH/NPS) – Vessel Failure: FOREVER program (KTH/NPS)

6.3 ROADMAP 3: MIXING AND STRATIFICATION IN RPV

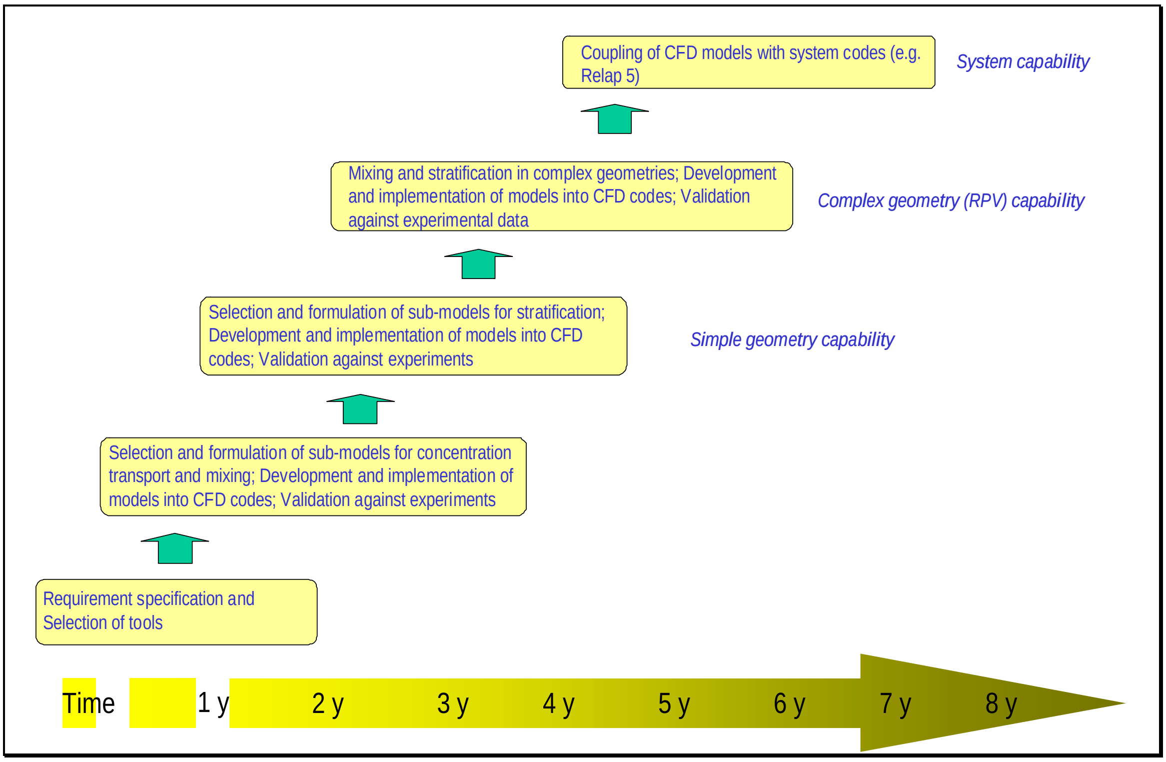


Figure 7: Roadmap 3, Mixing and stratification phenomena in RPV

Table 3: Summary of objectives, benefits and existing skills in the field of mixing and stratification in reactor pressure vessel

Lead:	Chalmers supported by Vattenfall Utveckling
Participants:	Chalmers, Vattenfall Utveckling AB, VTT, Westinghouse
Description:	– Includes macroscopic modelling with tools like RELAP combined with integrated and simultaneous CFD calculations for parts of the vessel volume. – Looks at thermal & boton mixing and stratification – Builds on and connects to planned project on ATWS (even ATWC) planned in Nordic co-operation (already funded by for example SKI).
Benefits:	– Develops state-of-the art CFD model of stratification and mixing – Enhances reactor safety assessments – Develops experimental skills
Research topics:	<u>Mixing</u> – Perform separate-effect experiments at Vattenfall Utveckling lab – Develop and validate a CFD model to predict mixing <u>Stratification</u> – Perform separate-effect experiments – Develop and validate CFD models to predict stratification in various geometries – Coupling Relap5 system code with CFD code to model both stratification and mixing
Existing know-how and skills:	– Experimental experience at Vattenfall Utveckling lab

7 SUMMARY AND CONCLUSIONS

A Nordic network for thermal-hydraulics and nuclear safety research was started. The idea of the network is to combine the resources of different research teams in order to carry out more ambitious and extensive research programs than would be possible for the individual teams. From the very beginning, the end users of the research results have been integrated to the network. Aim of the network is to benefit the partners involved in nuclear energy in the Nordic Countries (power companies, reactor vendors, safety regulators, research units).

First task within the project was to describe the resources (personnel, know-how, simulation tools, test facilities) of the various teams. Next step was to discuss with the end users about their research needs. Based on these steps, few most important research topics with defined goals were selected, and coarse road maps were prepared for reaching the targets. These road maps will be used as a starting point for planning the actual research projects in the future.

The organisation and workplan for the network were established. National coordinators were appointed, as well as contact persons in each participating organisation, whether research unit or end user. This organisation scheme is valid for the short-term operation of NOTNET when only Nordic organisations take part in the work. Later on, it is possible to enlarge the network e.g. within EC framework programme.

The network can now start preparing project proposals and searching funding for the first common resarch projects.

APPENDIX 1 DESCRIPTION OF VTT

VTT Technical Research Centre of Finland – a technology forerunner

VTT is an impartial, internationally respected and networked research centre. Today VTT is an expert organisation with a turnover of 230 million € and a staff of 3000. Its goal is to achieve practical benefits through high-quality research that brings added value for the customers. VTT’s major clients and co-operation partners are industrial and other companies and businesses, universities and research institutes. VTT produces new technologies in co-operation with domestic and foreign partners.

VTT’s research covers a broad range of technologies: information&communication, energy, industrial processes and systems, materials and manufacturing, biotechnology and life sciences, as well as building and transport. This interdisciplinary expertise available at VTT is a strong additional asset for innovative research. VTT also has proven experience in forming tailored partnerships with Finnish and foreign research organisations and companies.

VTT Nuclear – an efficient access to VTT’s comprehensive nuclear expertise

VTT Nuclear is the gateway to all nuclear energy research services at VTT. VTT Nuclear provides almost 200 dedicated nuclear experts. In addition, the customers have access to the expertise and networks of the entire VTT's staff and its modern experimental facilities and calculational tools.

Continuous development of new know-how and services

VTT has a central role in Finnish nationally coordinated research and technology development programmes: predominantly publicly funded programmes on nuclear power plant safety and nuclear waste management as well as industry driven programmes on plant life management and advanced reactors. In this way VTT maintains a comprehensive up-to-date know-how, computer code system and experimental facilities for the design, safety analysis and operational support of nuclear power plants and waste management facilities.

VTT complements and strengthens its own know-how also via extensive participation in international projects and networks within EU, OECD/NEA and IAEA as well as with bilateral collaboration with several foreign institutes and companies.

Recent research services and future challenges

VTT is the major technical support organisation for the utilities Fortum and TVO operating Finnish nuclear power plants, for the waste management company Posiva and for the safety authority STUK, and has also numerous contracts from foreign utility, industrial and regulatory customers.

In recent years VTT´s capabilities have been in extensive use in the major modernisation and power uprating programmes of the Finnish nuclear power plants and in the systematically proceeding implementation of the nuclear waste management programme. VTT has also participated in Finnish and international projects aimed at raising the safety level of Central and Eastern European nuclear installations.

In the forthcoming years, the construction and licensing of a new Finnish nuclear power plant unit, plant life extension of the existing plants, including modernisation of the I&C systems, improved fuel utilisation by higher burnup and the next steps of the waste management programme are our major challenges.

In addition to contract and collaborative R&D projects, VTT Nuclear offers delivery and tailoring of software products and experimental and measuring devices developed at VTT. Software tools include APROS process simulation environment, reactor physics and dynamics codes, computerised tools for emergency preparedness and code package FEFTRA for groundwater flow&transport modelling.

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APPENDIX 2 DESCRIPTION OF WESTINGHOUSE ATOM

Westinghouse Electric Sweden

Westinghouse in Sweden and Westinghouse Electric Company in the USA are wholly owned subsidiary companies of BNFL, British Nuclear Fuels plc. BNFL, with its 23,000 employees, has a goal to become the global leading nuclear company. Within this new company group, Westinghouse, with approximately 9,000 employees in the USA, Europe and Asia has the responsibility for the front end of the fuel cycle and is organised in three business units: Nuclear Fuel, Nuclear Services and Nuclear Plant Projects (Automation). Westinghouse in Sweden is responsible for the Boiling Water Reactor (BWR) technology within each Business Unit.

The Swedish company has built nine nuclear power plants in Sweden and two in Finland, based on its own, independent design. Westinghouse in Sweden today is a supplier of new BWR plants, nuclear fuel, plant upgrades including nuclear automation as well as nuclear services on the markets in Europe, the USA and Asia. The company in Sweden is a knowledge-based company with more than 40 years of successful experience with nuclear power. Approximately 800 employees in Västerås are engaged in nuclear activities.

The Westinghouse nuclear fuel facility in Västerås, Sweden is one of the most modern fuel manufacturing facilities in the world and is a vital part of the Company's fuel operations. The nuclear fuel facility fabricates Light Water Reactor fuel, BWR channels and BWR control rods.

Westinghouse in Sweden is the centre of the BWR technology and is constantly developing its products to keep high standards and meet customer requirements. A good fuel design has to take into account a number of requirements and must be an attractive compromise between these requirements, which often are contradictory:

- nuclear design with an optimal distribution of fossil material, moderator and neutron absorbers

- heat transfer and fluid dynamics

- transients and stability analyses including anticipated accidents scenarios

- strength of materials and material performance

This optimisation requires quite a lot of good ideas as well as knowledge within many different fields. As a support to these processes, the company has developed a considerable number of advanced computer programs to simulate the fuel performance in reactor conditions. In addition, the new designs are tested in different test rigs, including the FRIGG loop.

The FRIGG loop is currently one of the most advanced test facilities in the world for both BWR and Pressurised Water Reactor (PWR) fuel assemblies. The total power of the loop is 15 MW, which enables testing of the whole full-scale BWR assembly. The tests include the measurements of the dryout power, the pressure drop, the void fraction and the stability and transient characteristics of fuel assemblies. The void fraction can be measured using the Computerised Tomography, which produces an image of a slice in a test-section by traversing it with γ-rays.

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APPENDIX 3 DESCRIPTION OF KTH

Kungliga Tekniska Högskolan (KTH)

Royal Institute of Technology (RIT) or Kungliga Tekniska Högskolan (KTH, in Swedish), in Stockholm, Sweden is a technical university with first-class education and research. It provides one-third of Sweden's capacity for engineering studies and technical research at post-secondary level. RIT has about 11,900 students and 2900 employees and there are about 1300 active post-graduate students. RIT trains architects and engineers at Master's and Bachelor's level, as well as doctors and licentiates. RIT is organized in six schools and a college of applied engineering.

There are some 40 departments. Each possesses a wide and comprehensive scientific competence for research and undergraduate education. Most activities related to nuclear energy are performed in Department of Energy Technology, which is organized in six divisions, including Nuclear Power Safety, Nuclear Reactor Engineering and Center of Nuclear Technology.

In the Division of Nuclear Power Safety, the research program is directed towards resolution of the safety issues that are important to the Swedish nuclear power plants. In particular, experimental and analytical research are performed for the physical phenomena inherent in the scheme adopted in Sweden for management of the severe accidents. Additionally, research is performed on the safety of eastern European nuclear reactors with particular emphasis on the Ignalina plant in Lithuania.

The Nuclear Power Safety Division (NPS) at the Royal Institute of Technology (RIT) is headed by Prof. B.R. Sehgal, who coordinated the MVI, the ISARRP Projects and participated in the MFCI and CSC Projects of EU's 4th Framework Programme and also coordinated the ARVI Project and participated in the Projects, ECOSTAR, EUROCORE, XADS and TECLA. During the last 10 years, more than 250 publications have been made in Journals and conferences.

A large laboratory is available in which induction and resistance furnaces have been installed and two containments have been constructed in order to perform large scale experiments with high pressure and high temperature oxidic mixture melt material, interacting with water or structural materials (vessel). A Phillips 320keV X-Ray source and a Thomson 290 mm image intensifier assembled with high-speed camera become one-of-kind high-speed X-ray radiography system to perform continuous visualization measurements in opaque media. The multi-sensor void probe technology was developed to measure local void fraction. A number of data acquisition systems to acquire various measurement signals up to 2GHz sampling rate are available. A high-pressure system (up to 25 bar) was also developed to perform the FOREVER Vessel Failure experiments with high pressure. Recently, real-scale (in length) single channel lead-bismuth liquid loop was constructed to investigate thermal-hydraulics of liquid lead-bismuth. The laboratory has performed many large-scale experiments in the last more than 10 years in various domestic and international projects Analytical efforts have been focused on developing in-house computational fluid dynamic codes to investigate intensive multiphase phenomena which occur in various processes related to nuclear safety. Moreover, various system codes, e.g., RELAP5, CATHARE, TRAC (TRACE), MELCORE, etc., have been used to analyze the phenomena that we have studied.

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APPENDIX 3 DESCRIPTION OF KTH

The Nuclear Reactor Technology Division at KTH performs research and development within thermal-hydraulics of LWRs. Currently dryout and post-dryout measurements are performed in the high-pressure two-phase flow loop. The loop allows measurements of two-phase flows with operating pressure of up to 250 bars and with total heat power of 1MW. In addition to the high-pressure loop, the division carries out detailed measurements of air-water flow in Plexiglas mock-ups of LWR fuel assemblies. The goal of the research is to develop and validate closure relationships for two-fluid model of multiphase flows in fuel assemblies. The new models will use the Computational Fluid Dynamics (CFD) technology and will be formulated in a generic form, allowing implementation in any commercial or special-purpose CFD code.

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APPENDIX 4 DESCRIPTION OF LUT

LUT Lappeenranta University of Technology

Lappeenranta University of Technology is a university of technology and economics, founded in 1969. At the moment the university has some 4500 students, number of students is continuously growing, as well as the university and the campus area.

Mechanical Engineering, Electrical Engineering, Industrial Engineering Management, Information Technology, Energy Technology, Chemical Technology, and Environmental Technology are the technology areas of the university. The Business Administration Department contains Accounting, Finance, International Marketing, Management and Organisation, Business Law, Supply Management, Knowledge Management and Technology Research sections.

Department of Energy and Environmental Technology

Department of Energy and Environmental Technology consists of a Section of Energy Technology containing Laboratories of Power Plant Engineering, Computational Fluid Mechanics, Fluid Dynamics, Nuclear Engineering and Nuclear Safety Research Unit, and a Section of Environmental Technology containing Thermodynamics, Energy Economics and Environmental Engineering.

Nuclear Engineering Laboratory and Nuclear Safety Research Unit

Nuclear Engineering Laboratory and Nuclear Safety Research Unit work in a close cooperation, the Nuclear Engineering Laboratory is more concentrated to education of students, also postgraduate, and the Nuclear Research Unit mainly deals with experimental studies in thermal hydraulics.

Nuclear Safety Research Unit

Nuclear Safety Research Unit and its predecessors have carried out thermal hydraulic experiments related to nuclear plant safety. Earlier the research was more concentrated to study the coolability of the reactor core in the accident situations. The main facilities developed from the single pin rewetting test facility to an integral model of PWR plant primary circuit including the main safety systems.

Later the research projects have related to the studies of single safety systems, such as core catchers, condensation pool, scram tank, other passive systems etc.

Nearly 30 years of experience on thermal hydraulic experiments gives a good background to investigate plant specific thermal hydraulic problems and to design experimental set-ups and test procedures.

Currently the Nuclear Safety Research Unit can offer an infrastructure consisting power supply system up to 1 MW electric power, PACTEL facility, which can be used as a steam source, several pressure vessels and tanks as well as a large open water pool. Current data acquisition systems will be updated to better fulfil the needs of measuring fast condensation phenomena.

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Title	Final Report of the "Nordic Thermal-Hydraulic and Safety Network (NOTNET)" - Project
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Affiliation(s)	VTT Processes, Finland * Current affiliation TVO, Finland ** Current affiliation STUK, Finland
ISBN	87-7893-166-5 *Printed report*
Date	April 2005
Project	NKS_R_2004_35
No. of pages	37
No. of tables	3
No. of illustrations	7
No. of references	5
Abstract	A Nordic network for thermal-hydraulics and nuclear safety research was started. The idea of the network is to combine the resources of different research teams in order to carry out more ambitious and extensive research programs than would be possible for the individual teams. From the very beginning, the end users of the research results jave been integrated to the network. The aim of the network is to benefit the partners involved in nuclear energy in the Nordic countries (power companies, reactor vendors, safety regulators, and research units.
Key words	Thermal-hydraulics, nuclear safety, Nordic network, NOTNET

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