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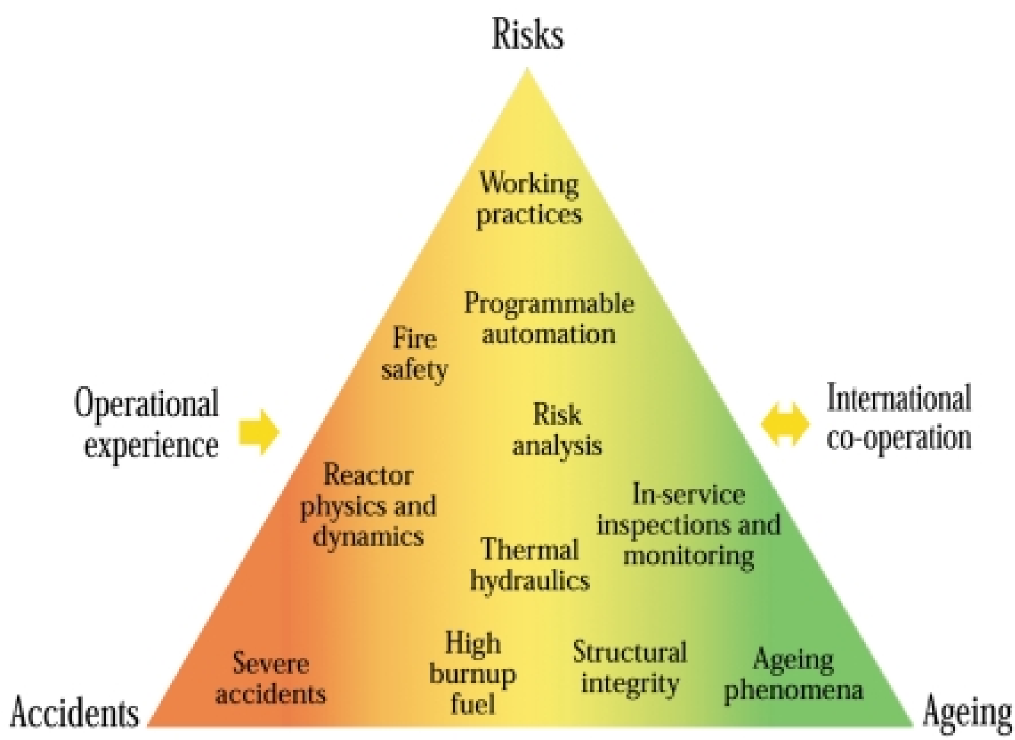
FINNUS

The Finnish Research Programme on Nuclear Power Plant Safety

Interim Report 1999 – August 2000



TECHNICAL RESEARCH CENTRE OF FINLAND ESPOO 2000

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The Finnish Research Programme on Nuclear Power Plant Safety
Interim Report 1999 – August 2000

Edited by

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VTT Energy

TECHNICAL RESEARCH CENTRE OF FINLAND
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Keywords FINNUS, nuclear power plants, reactor safety, materials, corrosion, ageing, accidents, thermal hydraulics, modelling, fire safety, risk analysis

Abstract

FINNUS (1999–2002) is the Finnish public research programme on nuclear power plant safety, launched and administrated by the Ministry of Trade and Industry (KTM). The programme concentrates on the themes of ageing, accidents and risks. The general objectives of the programme are to develop tools and practices for safety authorities and utilities, to provide a basis for safety-related decisions, to educate new nuclear energy experts and to promote technology and information transfer. The technical objectives of the programme are prepared in the guidance of the Radiation and Nuclear Safety Authority (STUK), but the views of the power companies are taken into consideration. Funding of the programme is mainly from public sources. The annual volume of the programme is about FIM 22 million (€ 3.6 million) and 30 person years. The research is coordinated and mainly conducted by the Technical Research Centre of Finland (VTT).

The effects of **ageing** on nuclear power plants are studied intensively, in order to evaluate safe remaining lifetime of the components and efficiency of the corrective measures. The programme concentrates on studies in material sciences of metallic structures, structural integrity and in-service inspection and monitoring methods. The **accident** theme concerns operational aspects of nuclear power plant safety. The issues of nuclear fuel behaviour, reactor physics and dynamics modelling, thermal-hydraulics and severe accidents are addressed under the theme. In the **risk** field attention is paid on one hand on advanced risk analysis methods and their applicability, and on the other hand, on the evaluation of fire risks, safety critical applications of software based technology, as well as human and organisational performance. The report summarises goals and results of the programme during the period 1999 – August 2000.

Preface

Organisation of public nuclear energy research in Finland as national programmes was started in 1989, launched by the Ministry of Trade and Industry (KTM). Since then these programmes have been carried out in the fields of operational aspects of safety, structural integrity and nuclear waste management. In parallel there have been running separate technology programmes on nuclear fusion, advanced light water reactor concepts and plant life management, funded partly by the National Technology Agency (Tekes).

KTM decided to continue the national research efforts on fission reactor safety in a single research programme after completion of the programmes on Reactor Safety (RETU 1995–1998) and Structural Integrity of Nuclear Power Plants (RATU2 1995–1998). The national advisory committee on nuclear energy, commissioned by KTM, made a general plan for the new programme and for its organisation (Advisory 1998), where the recommendations of the international evaluation of the RATU2 and RETU programmes and opinions of national expert panels were taken into account (Faidy & Hayns 1998, RETU 1998, RATU2 1998). The new programme, *The Finnish Research Programme on Nuclear Power Plant Safety (1999–2002) FINNUS [Kansallinen ydinvoimalaitosten turvallisuustutkimusohjelma],* concentrates on the themes of ageing, accidents and risks.

The Technical Research Centre of Finland (VTT) coordinates the FINNUS programme and also performs most of the research. The main funding sources are KTM, VTT, the Radiation and Nuclear Safety Authority (STUK), the Lappeenranta University of Technology (LTKK) and the power companies Teollisuuden Voima Oy (TVO) and Fortum Oyj. These parties are also represented in the steering group of the programme.

The execution of the programme is based on the general plan and annual plans prepared for KTM in cooperation with the regulatory body, the power companies and the research bodies. This Interim Report summarises general features of the FINNUS programme and the main goals and achievements in the period 1999 – August 2000. The report includes selected, more detailed results in the form of presentations in the FINNUS Mid Term Seminar Oct. 31st – Nov. 1st, 2000, Otaniemi. Statistical information on publications, contact lists and academic degrees awarded are also included.

The report has been prepared by the programme coordination unit, in cooperation with the project leaders and the programme staff.

Contents

1. Introduction

About one third of electricity generated in Finland is nuclear. High operational reliability of the nuclear power plants has led to low production costs. In order to maintain nuclear power as a major source of energy it is very important to ensure high level of nuclear safety. This requires continuous effort on research in various fields of technology and human aspects, in order to take into account modernisation and upgrading of plant processes, implementation of new techniques, changing production goals and renewing safety requirements. Continuous high level of safety is a prerequisite for public acceptance of nuclear energy and for maintaining of future nuclear option.

1.1 Nuclear energy research in Finland

The total volume of nuclear energy related R&D efforts in Finland is currently about FIM 160 million/a, or € 27 million/a, figure 1. The power companies fund directly more than half of the total volume and the public sector about one third. Half of the total volume is spent for nuclear waste management issues, mainly conducted by the power companies. Nearly 40 % of the resources are used for reactor safety, out of which about half is indicated for fully or partly public research programmes.

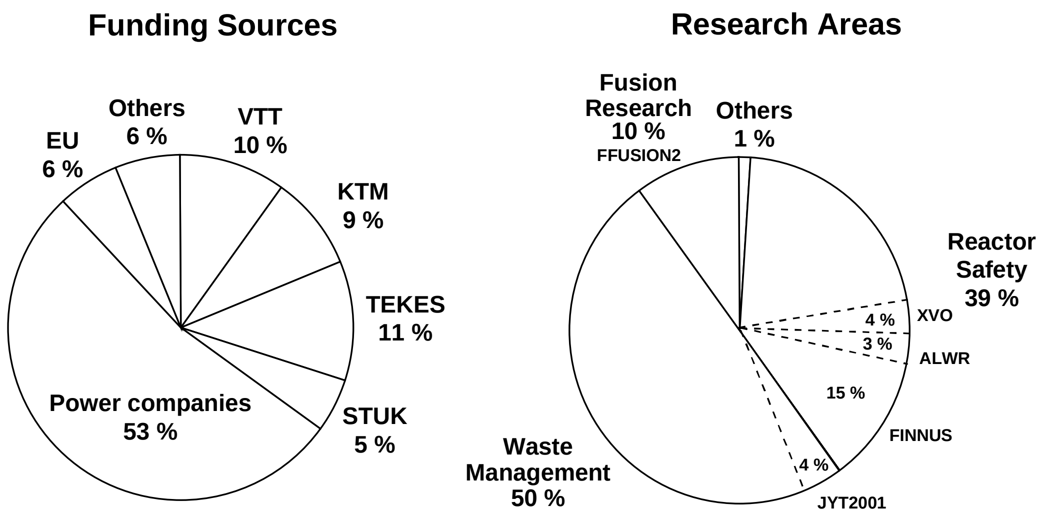

Figure. 1. Resources of nuclear energy research in Finland in 1999. The public funding comes from the Ministry of Trade and Industry (KTM), the Technical Research Centre of Finland (VTT), the National Technology Agency (Tekes) and from the Radiation and Nuclear Safety Authority (STUK).

Currently the largest of the public programmes is the Finnish Research Programme on Nuclear Power Plant Safety FINNUS (1999–2002), that concentrates on nuclear reactor safety related issues of the existing power plants. The Advanced Light Water Reactor programme ALWR deals with possible future solutions of nuclear power generation and the Plant Life Management Programme XVO is directed towards plant specific ageing problems with particular support from the power companies. These programmes are mainly conducted at the various research units of the Technical Research Centre of Finland (VTT). Universities also contribute to these programmes.

1.2 Organisation and general goals of the FINNUS programme

In the planning of the new research period it was recognised that a focused and result oriented research programme is necessary, where various fields of nuclear safety research are pooled and where strategic planning of resources is facilitated. Thus, the previously separated fields of structural and operational safety research were combined into the Finnish Research Programme on Nuclear Power Plant Safety FINNUS (1999–2002). The Ministry of Trade and Industry (KTM) launched the programme and administrates it. In order to guarantee independent know-how and research resources for the public sector, the needs of Radiation and Nuclear Safety Authority (STUK) were dominant in the planning of the programme. This aspect also reflects to the funding and steering structure of the programme. In order to combine the limited national resources, the Finnish power companies also contribute to the national programme by offering resources to specific tasks and and by participating in the steering and reference groups. The research is mainly performed by VTT, that also coordinates execution of the programme (FINNUS 1998, FINNUS 2000a, FINNUS 2000b).

The general objectives of the FINNUS programme in the long term are to

- develop tools and practices for safety authorities and utilities,
- provide a basis for safety-related decisions,
- educate new nuclear energy experts and
- promote technology and information transfer.

<table>
<tr><th colspan="2">FINNUS (1999–2002) briefly:</th></tr>
<tr><td>Research fields:
NPP ageing, accidents and risks

Total volume (1999–2002):
FIM 90 mill. / € 15 mill.
120 person-years

Funding in 2000 (FIM / € mill):
KTM: 9.1 / 1.53
VTT: 6.6 / 1.11
STUK: 3.7 / 0.62
Utilities: 0.9 / 0.15
Other: 1.4 / 0.23</td>
<td>Research Partners:
VTT Energy,
VTT Manufacturing Technology,
VTT Automation,
VTT Building Technology,
VTT Chemical Technology,
Lappeenranta University of Technology

Coordination:
VTT Energy</td></tr>
<tr><td colspan="2">http://www.vtt.fi/ene/tutkimus/finnus/index.html</td></tr>
</table>

In the organisation of steering and support functions of the programme, strong coupling is ensured between the funding bodies, research organisations and the end users of the results, both at the nuclear safety regulator and the power companies. The main duty of the steering group is to supervise the overall performance of the programme. The steering group sets up reference groups for the research projects and nominates strategic planning groups on the main themes. The project reference groups contribute to the planning of the projects, e.g. by suggesting new research topics. They advise the projects and evaluate the results against the plans. An important task of the reference groups is to communicate between the research groups and the users of the results. The strategic groups evaluate priorities and discuss the challenges of the whole programme against national needs. The steering group and the project reference groups typically meet 3–4 times a year, while the strategic planning groups meet roughly twice during the programme. Organisation scheme of the FINNUS programme is shown in figure 2.

A total of 11 research projects are under way in the field of ageing, accidents and risks as indicated in figure 3 and table 1. There are natural connections between the projects, and one objective of the programme is to strengthen these links. All the projects have been planned for the four years period, but detailed objectives vary during execution of the programme. Chapters 2–4 summarise the main goals and results of the projects in the first period of the programme. Capter 5 consists of a selection of more detailed research results.

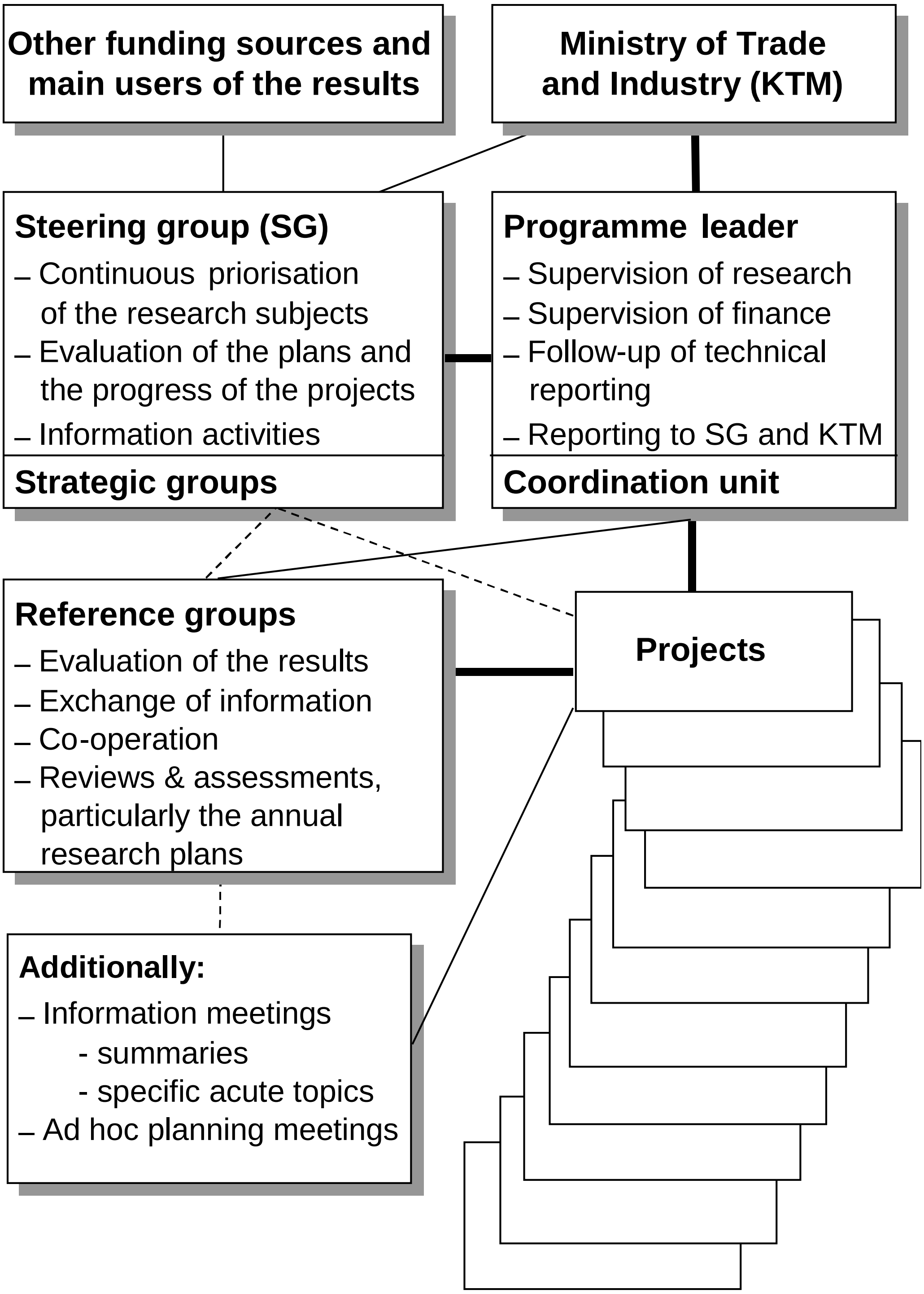

Figure 2. Organisation of the FINNUS programme.

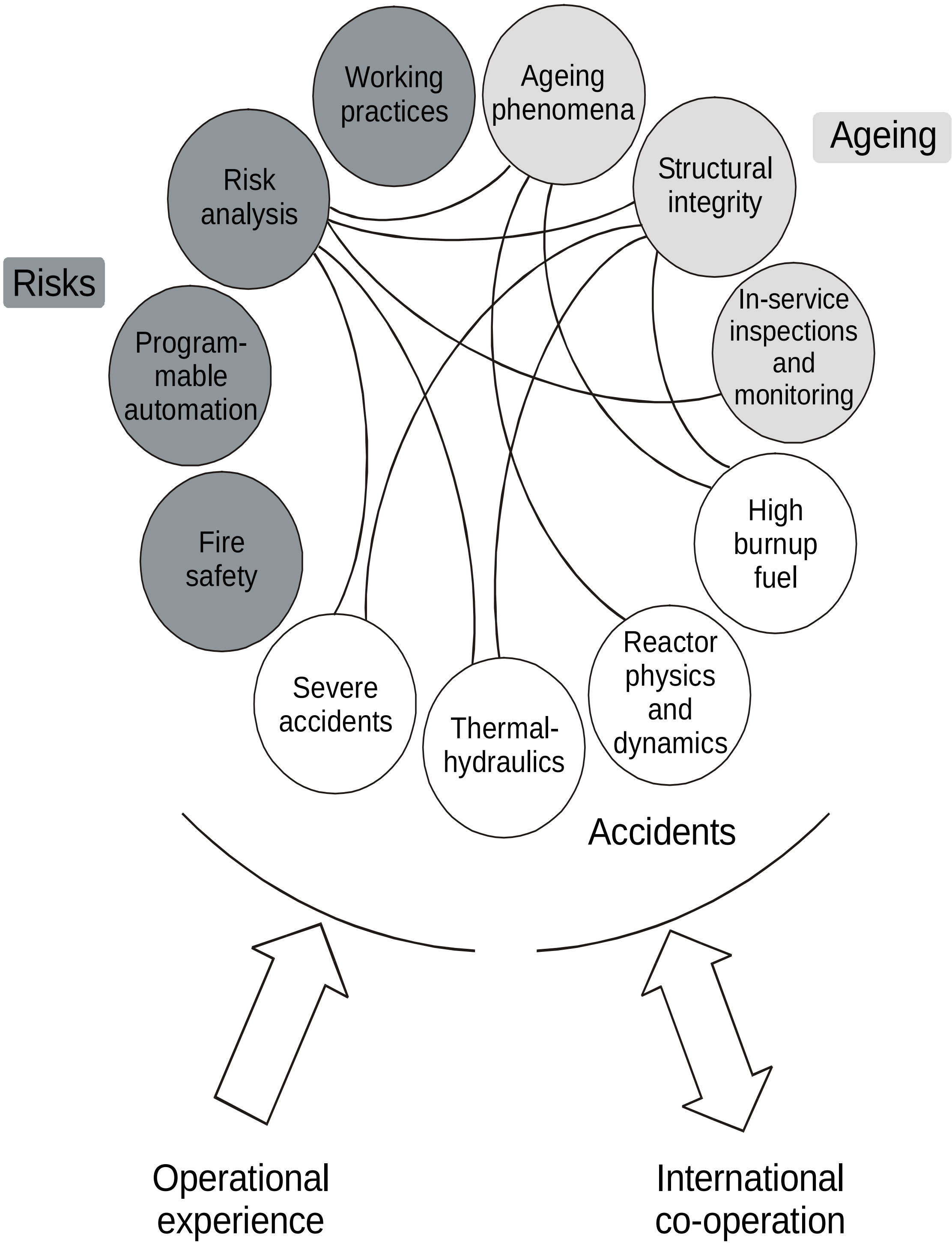

Figure 3. The 11 research projects under the three themes of the FINNUS programme. The projects mainly associated with a certain theme are strongly intercoupled. Couplings to be strengthened between the themes are indicated.

1.3 Contents of the FINNUS programme

The research objectives of the programme may be classified under three themes, **ageing**, **accidents** and **risks**.

Table 1. Volume and funding of the FINNUS projects.

Main theme / Project (Acronym)	Research organisations	Volume [person years] 1999	2000 plan	Costs [FIM mill.] 1999	2000 plan
Ageing					
Ageing phenomena (AGE)	VTT Manuf. T. VTT Chem. T.	5.3	3.0	3.9	3.3
Structural integrity (STIN)	VTT Manuf. T. VTT Energy	2.7	3.4	2.0	3.0
In-service inspections and monitoring (INSMO)	VTT Manuf. T. VTT Building T.	1.2	1.2	1.0	0.9
Accidents					
Behaviour of high burnup fuel in accidents (KOTO)	VTT Energy	2.4	1.6	1.3	1.0
Reactor physics and dynam. (READY)	VTT Energy	6.7	4.7	3.3	2.9
Thermal hydraulic experiments and code validation (TOKE)	VTT Energy Lappeenranta Univ. of Tech.	4.4	3.2	3.1	2.5
Modelling and simulant experiments of severe accident phenom. (MOSES)	VTT Energy VTT Manuf. Technology	3.3	2.5	2.6	2.5
Risks					
Fire safety research (FISRE)	VTT Building Technology	1.0	1.9	0.6	0.9
Programmable automation system safety integrity assessment (PASSI)	VTT Automation	1.7	2.1	0.8	1.3
Methods for risk analysis (METRI)	VTT Automation	2.3	2.5	1.5	1.5
Working practices and safety culture in NPP operations (WOPS)	VTT Automation	2.2	2.2	1.1	1.4
Administration and information of the research programme (HALTI)	VTT Energy	0.9	0.7	0.7	0.6
Total		34.1	29.0	21.9	21.8

Ageing

The ageing field covers material sciences of the metallic structures in a nuclear power plant, structural integrity studies and in-service inspection and monitoring methods. Three research projects, AGE, STIN and INSMO mainly concentrate on these issues.

Ageing of mechanical components, fuel and constructions is one of the main factors limiting remaining lifetime of nuclear power plants. Identification of critical components, ageing mechanisms and changes in in-service properties is crucial in order to develop preventive actions as well as remedial measures and to evaluate the remaining component lifetimes. In the Ageing Phenomena project (AGE), methods for determining in-service properties are developed and applied for aged materials and components. The aim is also to model the effects of water environment and irradiation exposure on the ageing phenomena and particularly, the effects of re-irradiation after annealing of pressure vessel material.

The ageing of equipment and structures can be computationally predicted and the safety margins in respect to failure during the service life can be assessed using numerical methods. The needed material properties should be characterised reliably using preferably small specimens. Specific methods are needed for modelling of material inhomogenities and local loading effects, such as impacts and thermal stratification. The main objectives of the Structural Integrity project (STIN) are creation of verified experimental and computational methods and also verification of the existing methods for assessing the remaining lifetime of components and their ability to withstand possible accidents.

In the In-service Inspections and Monitoring project (INSMO), non-destructive testing (NDT) methods are developed, evaluated and applied to examine the structural integrity of the critical components. The reliability of in-service inspections is improved by applying advanced inspection techniques and new analysis methods. The issues concerning qualification of inspection methods are seen as a very important area, and different research efforts are directed to develop practices according to the quidelines of the European qualification methodology. On-line monitoring techniques are considered for acquisition of structural integrity data during the operation of plant as well as non-destructive methods to measure material properties. The techniques for inspections of reinforced concrete structures are seen as an important area of plant condition monitoring.

Accidents

The accident field covers fuel research, reactor physics and dynamics, experimental and calculational thermal-hydraulics and severe accidents, that are studied in the KOTO, READY, TOKE and MOSES projects.

In the Behaviour of High Burnup Fuel in Accidents project (KOTO), calculation tools for fuel rod thermal-mechanical performance and safety assessments in steady state and transient conditions are qualified to cover higher burnup and the fuel designs used in Finland. The safety authorities in different countries widely share the opinion that the data base of the acceptance criteria for design basis accidents may be insufficient, in relation to the burnup targets currently desired in search for better fuel efficiency and less fuel waste. To understand the phenomena in more detail and to support their modelling, extensive research is under way, including internationally sponsored reactor experiments and out-of-pile testing, in which Finnish participation is sought.

A computer code system and competence for carrying out reactor physics and dynamics calculations needed in Finland has been developed at VTT in the Reactor Physics and Dynamics project (READY) and its predecessors. In reactor physics the main objective is to ensure the reliable use of the wide range code system in order to generate nuclear data for new fuel types and for transient and accident analyses, as well as to carry out flux, dose rate and criticality safety studies. In reactor dynamics the objectives are to complement and validate the calculation system for complex reactivity accident and stability studies. The thermal hydraulic models of the dynamics codes are improved by developing and taking into use new methods and models. The project aims at on-the-job training of new experts. It also contributes to VVER safety improvements in international co-operation.

In the Thermal-hydraulic Experiments and Code Validation project (TOKE), the objective is to produce thermal-hydraulic data for developing more accurate safety analysis tools and to enhance understanding of abnormal situations in nuclear reactors. The project contains three parts: VVER related tests with the PACTEL test loop, separate effect tests and validation of thermal-hydraulic computer codes. PACTEL is a volumetrically scaled thermal-hydraulic model of the Loviisa power plant. The aim is to expand the scope of VVER experiments and to assure that there is a validated computer code for safety analysis. The computer code validation is carried out in close co-operation with the experimental programme. A further objective is to find out boundary conditions for assessment of the effects of a power plant lifetime, such as thermal fatigue of the pipes.

The safety of the Finnish nuclear power plants against severe accidents has been enhanced with several plant modifications and extensive research during the past decade. However, significant uncertainties still remain in some areas that are important to the Finnish NPPs, such as coolability of core debris, failure of the pressure vessel and fission product behaviour, particularly in the reactor coolant system. The Modelling and Simulant Experiments of Severe Accident Phenomena project (MOSES) focuses on reducing uncertainties and getting better insights on pressure vessel lower head response to core melt attack, coolability of debris beds, chemistry of fission product iodine and containment thermal hydraulic loads.

Risks

The risk field covers topics on fire safety FISRE, programmable automation PASSI, methods of risk analysis METRI and human factors WOPS.

The Fire Safety Research project (FISRE) studies influence of smoke and heat from fires on electronics and programmed logic circuits. Fire risk analysis is developed further towards living probabilistic safety assessment (PSA). First very simple fire models are used for sensitivity types of studies with an emphasis on plausible probability distributions of input variables. Guided by the obtained results more sophisticated methods are developed later for sensitive parts of risk analysis link. Likewise the reliability of active fire protection measures, such as sprinklers and alarming systems, is further studied.

In the field of safety and reliability assessment of programmable automation systems new and more cost-effective methods, tools and practices are developed and tested in order to support the authorities and utilities in solving the licensing problems of software-based safety critical automation systems. The work of the Programmable Automation System Safety Assessment project (PASSI) is partly focused on the development of reliability assessment methods and tools, partly on experimental application studies aiming at the recognition of further development needs and at the creation of a proper base for justification of safety claims. A pre-study on ageing of I&C equipment in nuclear power plants is also included in the project.

Probabilistic safety assessment (PSA) is increasingly used in safety management and regulation of NPPs. In the Methods of Risk Analysis project (METRI), risk informed decision making methods are promoted and licensing practices, as well as methods for risk importance and uncertainty are developed. Many problems of the risk informed decision making are due to balancing between probabilistic and other criteria, uncertainties in PSA results, the licensing complexity and some specific questions e.g. common cause failures and human reliability. The main objectives of the project are: to make progress in risk informed decision making methods and bring them available for end users; develop licensing practices in problematic areas; develop methods for importance & uncertainty analysis; develop skills in nuclear risk analysis, assure the competence transfer to the new generation and to contribute to international perspectives of quantitative risk analysis.

The Working Practices and Safety Culture in Nuclear Power Plant Operations project (WOPS) aims to enhance human-machine interface, competences and training of operating staff and development of organizational practices. A method for the analysis of working practices has been developed in earlier studies. It has been applied as a tool for the control room operators' simulator training. The present project developes further applications. In integration with probabilistic risk assessment the method is applied in the analysis of the safety effects of different types of control room information presentation. It will also be used in studying the role of emotional and cultural factors in the organisation of situated actions in every-day work. These topics require a further theoretical elaboration of the method.

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2. Ageing: Main goals and results

2.1 Materials Ageing Phenomena in NPP's (AGE)

Main work on ageing is oriented to extend the operating lifetime of the nuclear power plants. Technically plant life extension is established and described since mid 80's. However, recurrently new degradation mechanisms have been discovered in aged materials and thus, these lately discovered ageing mechanisms are not included in the life management plans. The new ageing mechanisms of components exposed to reactor environments are studies in most of the countries using nuclear energy. Examples of these new ageing related degradation mechanisms are:

- Intergranular stress corrosion cracking (IGSCC) observed in stabilised or in cold deformed, non sensitised austenitic stainless steels.
- Irradiation assisted stress corrosion cracking (IASCC) observed in austenitic stainless steels even in highly reducing environments like in PWR's.
- Intergranular or interdendritic cracking of nickel base alloys.
- Dynamic strain ageing of nitrogen alloyed ferritic steels as well as low carbon austenitic stainless steels.

Additionally, remedial actions taken in the nuclear power plants to prevent or mitigate the degradation caused by the ageing mechanisms have regenerated new open questions:

- Effects of new water chemistries on fuel cladding materials.
- Annealing and re-embrittlement of irradiated reactor pressure vessel steels.
- Water chemistry transients effect on cracking of pressure boundary materials during applied modified water chemistry.
- Alkaline water chemistry in the secondary side and the SCC of stainless steel steam generator tubing under pile ups.

The significance of materials science for effective plant life management is clear: the mechanisms underlying in materials degradation must be understood in order to define and quantify suitable corrective and preventative measures. Material technology contributes significantly both to the scientific background of fundamental research as well as to everyday engineering work during all phases of life time of power plants. Most of the work is focused on the passive structures and components, that have long life and are very difficult to repair or replace.

The major concern of all programs related to life extension of LWR components is the behaviour of systems and components whose integrity is essential for normal operation of a plant (IWG-LMNPP-98/2). These systems and components are needed as a part of the coolant pressure boundary or they are necessary to shut down the reactor, to maintain a safe condition or to minimise on- and off-site exposures. They should be able to serve for the design life of the plant despite some ageing degradation. Therefore, understanding the ageing mechanisms and monitoring of the degradation processes are of key importance in safe plant operation. Smaller, less expensive items, such as sensors, certain pumps, valves etc. are serviced, repaired or replaced on a fairly frequent basis. Their correct functioning is also of importance, and probabilistic risk assessments can be used to predict their optimum service of replacement time.

Main objectives

In the FINNUS program the targets and goals for the ageing related materials research work have been defined along the lines of international programs (Advisory 1998). These were:

- Understanding and modelling of the environment assisted degradation and corrosion of main component materials in various LWR water chemistries.
- Definition of the re-embrittlement rate and mechanism of the reactor pressure vessel steels in Loviisa after annealing heat treatment.
- Neutron irradiation induced degradation of reactor internal materials.
- Thermal ageing of cast austenitic stainless steels and weld metals during long term exposure to reactor operational temperatures.
- Wear of materials.
- Concrete ageing.
- Ageing of electric components and cables.
- Maintenance and repair methods.

The AGE project, divided into four technical materials related sub-programmes, was initiated to provide experimental data specific to Finnish plants, to establish understanding of ageing mechanisms involved and to transmit information from the international programmes related to ageing. Research items dealing with concrete, electric components or cables was not initiated within theAGE project, but are to some extent covered by other FINNUS projects.

Main results in 1999–2000

Environmentally assisted cracking of nuclear materials (ENVI)

It has not been possible to take into consideration all deterioration mechanisms, combinations of thermal, chemical and mechanical loads in the original plant design, because their existence was not realised at that time. Autoclave testing in laboratories has been developed to carry out tests applying simultaneously different loading parameters and chemical environments at elevated temperatures and pressures. Autoclave testing, however, has not been able to provide enough data for statistical evaluations in the case of EAC phenomena, contrary to fracture mechanics testing in air. Development of both loading and testing equipment, capable of testing several specimens simultaneously while keeping environmental parameters roughly constant, is a prerequisite for the application of statistical analysis methods, which is needed both for crack initiation and growth rate evaluations.

Fatigue design curves of nuclear components are based on fatigue tests that do not cover all factors known to influence the fatigue life, but include a safety factor covering uncertainty of these aspects. One clear deficiency is that the design curves do not address either the effects of ageing induced changes in the material properties, or the effects of reactor coolant, or the effects of oxidation on fatigue initiation.

Historically, autoclave fatigue testing has focused on crack growth rate measurements using pre-cracked specimens. However, a clearly expressed need for fatigue crack initiation measurements in reactor environments has been indicated by the obvious thermal fatigue or stratification failures observed at the plants. In order to study the crack initiation in highly stressed regions and to re-assess the original fatigue life calculations, expensive and time consuming crack initiation measurements in autoclaves are needed. Additionally, on-line monitoring of thermal cycles conducted at many plants, as part of ageing related monitoring programs, requires information of real initiation times in real environments to become useful.

As a response to this need, development of a bellows loaded fatigue rig, which is able to load several specimens simultaneously, was initiated. The prototype to be used under load controlled fatigue tests in an autoclave has been planned and built. Demonstrations in air at room temperature, as well as at high temperature and high pressure have been successfully performed. The bellows loading technology, previously used for other types of testing in LWR autoclaves, is also suitable for fatigue testing. The prototype axial fatigue unit provides controlled reversed loading and has sufficient load train rigidity and concentricity (Marquis et al. 1999). The fatigue unit is able to maintain alignment of the specimen almost as would be required from conventional materials testing load frames. The degree of bending is 11%, which qualifies as a class 20 alignment, but the bending is primarily static and the dynamic component is only 2%. The degree of alignment is well suitable for LCF and HCF testing of ductile materials, such as stainless steels. The strain controlled fatigue test operates without difficulty for 15600 fatigue cycles and the results have been in good agreement with the data from VTT and other labs that have generated data using more conventional test systems. The required load capacity of the fatigue unit can be achieved. The maximum cyclic frequency attainable by the prototype unit is 0.5 Hz and was limited by air flow of the bellows. Future work should allow fatigue test frequencies up to 1 Hz, if necessary. Four of the prototype units can be simultaneously placed in a single autoclave and further development may permit simultaneous testing of six specimens. The current bellows control system and the fatigue unit performed well for either stress controlled or strain controlled loading. The construction of the first autoclave testing rig with four positions is going on and the first material tests will be carried out and reported as part of AGE during the latter half of the project.

The behaviour of oxide films with regard to their role in activity build-up and different corrosion phenomena in nuclear power plants (OXI)

During ageing it is crucial to know changes in the properties of oxide films that are formed on construction materials during plant operation and possibly due to irradiation. This knowledge bridges applied water chemistry, environmentally assisted cracking and irradiation assisted stress corrosion cracking phenomena (EAC and IASCC), as well as other forms of corrosion. Additionally, history of water chemistry influences activity build-up and thus, maintenance of the plants. On the other hand detailed knowledge of oxide properties is required for material selection for replacements, pre-filming before power use in primary circuits, water chemistry practice during various stages of fuel cycle and for possible decontamination procedures at the plants. This sub-program aims at predicting the influence of operational conditions of nuclear power plants on occupational dose rates and on the corrosion susceptibility of the plant components when exposed to coolant water. This requires modelling of the processes taking place in the oxide films formed on component surfaces.

The sub-program has been divided into four tasks. The first task focuses on modelling the adsorption of radioactive species from the coolant on the outermost oxide surface. While considering the competitive interaction of various dissolved ions with the oxide film, the first step is considered to be adsorption. It is for instance possible that the influence of zinc ions on Co-60 incorporation is due to the competition between zinc and cobalt for the adsorption site. Model calculations have already demonstrated that the retarding effect of zinc in the coolant on the adsorption of cobalt on iron oxide can be theoretically predicted on the basis of the surface complexation approach.

The second task focuses on the transport of radioactive species in the outer, porous part of the oxide film. The first step has been to model the ionic transport in a BWR type coolant. The results have made it possible to estimate driving forces such as potential drops within the pores as a function of oxide properties.

The third task focuses on the compact, inner part of the oxide film. For oxidation of the construction material to proceed, species have to be transported through the existing oxide film. The inner part is likely to determine the rate of the whole oxidation. The mixed-conduction model (MCM) introduced earlier for room-temperature films has been successfully extended to high-temperature oxide films on Fe-Cr model alloys, resembling closely construction materials used in LWR's. The MCM makes it possible to explain qualitatively the effect of corrosion potential (in other words, the amount of oxygen or other oxidising agents) and of the film structure on the distribution of ionic defects in the film. It has also been found to account for the higher ionic conductivity of the film at elevated temperatures when compared to ambient temperatures (Beverskog et al. 2000). A model for the behaviour of oxide films on Ni-Cr alloys in nearly-neutral conditions at room-temperature has also been proposed, and measurements for Ni-Cr alloys at 200°C have been completed.

The experimental data used as input for the modelling work carried out in this project has required the development of new electrochemical techniques. Conventional electrochemical arrangements are usually not suitable for the high-temperature, high-pressure environments and poorly conductive media encountered in NPP cooling systems. Accordingly, a controlled-distance electrochemistry (CDE) arrangement has been introduced, and its applicability to thin-layer electrochemical, contact electric resistance and contact electric impedance measurements has been verified.

The results obtained in this subprogram of the AGE project will be reported as in-kind information for the EPRI / Co-operative Irradiation Assisted Stress Corrosion Cracking Research Program (CIR II) according to the co-operative agreement signed between EPRI and VTT in September 2000. CIR II program is among the other things planning to characterise factors affecting crack initiation through the interface between irradiated material and the coolant, i.e., oxide film. The interesting questions in the CIR II program are the effects of:

- Increased yield strength caused by irradiation and ageing on electronic properties of passive film,
- Material composition on passive film properties in connection to radiation induced grain boundary segregation (RIS),
- Corrosion potentials, internal oxidation and catalyst and radiolysis reactions on IASCC.

The effect of irradiation (I), annealing (IA) and re-irradiation (IAI) on material properties and the repair methods (EMBRI)

Many of the VVER 440 reactor pressure vessels, including Loviisa 1, have been annealed as a plant life extension measure due to their excessive irradiation embrittlement. The annealing, however, has not been considered in the plant design. Thus, the surveillance programs do not include enough full size specimens to cover additional material conditions, arising from ageing and conducted annealing. Studies characterising the use of sub-size and/or reconstituted specimens in IA and IAI conditions are currently going on. Research on the theoretical bases of material behaviour in annealing and the effects of re-irradiation in the micro-mechanistic and macroscopic level is currently going on. The estimation of the relevance and reliability of different toughness measures/tests for material ageing (I-, IA- and IAI-conditions) is also an essential object of the sub-program EMBRI.

Experimental tests have been carried out to create data base for the I- and IA-material conditions within the IAEA VVER 440 round robin programme in co-operation with OECD Halden project. For the moment, the data base includes CH-V and cleavage initiation fracture toughness data measured with specimens of different sizes. The data is used for creating and supplementing property-property and specimen size correlations developed by VTT. Based on the obtained results identification of the acceptable lower limit of the specimen size for fracture toughness tests has been discussed in many published papers (Valo et al. 1999). Reanalyses of existing data, supplementary tests with irradiated IAEA CRP materials, irradiation shift estimations and clarification of the inconsistent lower shelf values are to be conducted in order to further verify the small specimen reconstitution technique.

In order to understand and model the effects of chemical composition on the irradiation embrittlement a large model alloy (33 alloys) test program has been carried out in co-operation with JRC Petten. Data base for analysing the effects of key impurities, i.e. P, Cu and Ni, on embrittlement has been established at VTT using irradiated sub-size specimens (Valo 2000). As a continuation some of the model alloys were annealed and inserted for re-irradiation in the core of Loviisa plant. Micro-mechanism studies using the Field Emission Gun Scanning Electron Microscope (FEGSTEM) to find out the role of phosphorus in embrittlement of the irradiated model alloys will be initiated during the year 2000 and continued later by using the re-irradiated samples.

Fuel cladding ageing (FUELI)

Changes in reactor water chemistry have been introduced as a mitigation method to prevent stress corrosion cracking (SCC) in stainless steels. Hydrogen dosing, noble metal dosing or zinc injection have been shown to affect the SCC growth rates as well as on he activity build-up. However, the evaluation of all effects of these new water chemistry approaches and transitions on the fuel cladding materials requires more information. The trend towards higher fuel burn-up in light water reactors imposes a demand for better corrosion resistance of the zirconium alloy cladding tube materials. A proper understanding of the corrosion behaviour of zirconium alloys in light water reactor coolant conditions calls for fast and reliable in-situ techniques to investigate the properties of the oxide films formed on their surfaces. The starting point in AGE FUELI work has been the use of electrochemical measuring techniques which can be used at high temperature aqueous environments instead of the traditional long term weight gain tests.

The FUELI sub-program has been dedicated to test and verify the applicability of the controlled-distance electrochemistry (CDE) arrangement to study the corrosion properties of the fuel cladding materials in simulated reactor environments. The experiments showed that the CDE arrangement is suitable for the solid contact measurements, i.e., contact electric resistance (CER) and impedance (CEI) measurements of Zircaloy-2 specimens used in this experiment. However, the tubular form of the specimens impeded the electrochemical measurements in a thin-layer geometry.

The CER technique was found to provide information of the dc resistance of the Zircaloy-2 oxide film during the initial oxidation stage. The CEI measurements, from their part, provided information of the film behaviour at longer exposure times. The results suggest that the transport process taking place in the oxide layer is likely to control the rate of oxidation of the material and it can be associated with transfer of oxygen species in the film. Based on a preliminary semiquantitative analysis of the results, the value of the diffusion coefficient D_i of oxygen ions or vacancies in the film was assessed (Bojinov et al. 2000). In the future this type of results will probably be used to predict the growth rates of zirconium alloy oxides and to compare the corrosion performance of different cladding materials under various water chemistries. The initiated work will be continued to further qualify the experimental arrangement.

International co-operation

There are several relevant organisations working in ageing issues. The national safety authorities, the research institutions, the utilities and the owners groups, and international organisations are developing rules and guidelines to guarantee the safe operation of the plants. International ageing management programmes addressing nuclear power plants structures, systems and components have been conducted under the auspices of different International Organisations, IAEA, OECD/NEA, EPRI and the EU (Ramirez et al. 1997).

Within the AGE project the research on materials ageing phenomena in NPPs is conducted in close co-operation with the international programs. The research aimed at understanding the mechanisms of ageing and the consequent degradation is the main goal of the FINNUS AGE project. Thermal degradation, corrosion, fatigue, radiation embrittlement of construction materials and all their combinations have been major subjects of concern.

Applications

The fracture mechanical testing methods to be conducted for aged materials in combination with the environmental parameters for aged materials makes it possible to evaluate material specific changes due to exposure in reactor environments. This is needed for to apply NDE at right locations and for maintenance efforts conducted at right time.

Note: A special report on the AGE project in chapter 5.1.

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2.2 Structural integrity (STIN)

The ageing of equipment and structures can be computationally assessed by showing that there is high enough safety margin in respect to failure mechanisms during the service life. The needed material properties should be reliably characterised using preferably small specimens. The existence of material inhomogenities in weld joints and in other discontinuities of geometry and material properties gives rise to specific demands on the modelling of the fracture mechanisms. Specific methods are needed also for modelling of local loading effects such as impacts and thermal stratification.

The target of the structural integrity project is creation of verified experimental and computational methods and also verification of the existing methods for assessing the remaining lifetime of components and their ability to withstand the possible accident situations.

Main objectives in 1999–2000

The main objectives of the structural integrity project are to develop methods for reliable material characterisation using small specimens, to participate and follow-up of the projects in this area in the international networks and to develop experimentally validated methods for the analysis of piping under thermal stratification and water hammer loads. These objectives are realised by developing and validating the testing and analysis methods for the direct determination of fracture toughness and generation of fracture toughness data with miniature size specimens. Also a correlation for determining the fracture toughness from Charpy-V data will be completed and a constraint correction to the Master Curve approach (Wallin 1999) will be developed. A study will be started in order to find out quantitative estimates for the constraint correction. A test programme will be planned and started to verify the correction method experimentally. The relationship of the crack initiation and crack arrest fracture toughness characteristic of various structural steels is studied.

The loading due to water flow and stratification is determined by computational fluid dynamics. Experimental test realised in the PACTEL-system is analysed. Computational results will be verified against measurements. Structural analyses will be carried out by using the loading transients that are calculated with a computational fluid dynamics (CFD) code. Computational results will be verified against measurements.

The micromechanical computational analysis methods are developed and verified. Numerical models are needed to assess fracture resistance determination with small specimens using advanced material models. Numerical methods are also applied to extend the micromechanical analysis methods to practical (mismatch) loading situations and to consider zones of varying material properties at welds. The Master Curve concept is integrated to numerical local approach modelling.

Analyses of a reinforced concrete wall under detonation conditions is continued. The structural analysis model will be improved based on the preliminary studies. The final model will be analysed by using more realistic loading transients. Structural integrity assessments for a pipe inlet and other structures will be performed. Loading transient is assessed by using a three-dimensional computation code.

The computer codes are verified by attending international comparison studies and networks. In the year 2000 the NESC 2 project (EU network for the evaluation of steel components) is participated putting the main focus on the possible new projects in the NESC network dealing especially with piping integrity.

Main results in 1999–2000

Test results indicate that miniature size specimens may underestimate large specimen fracture toughness contrary to fracture mechanics theory. Co-operation with University of Illinois regarding constraint correction to the Master Curve has started. A relationship between the Master Curve fracture toughness and the T-stress has been developed (Wallin 2000). This enables the application of the Master Curve Method also with low constraint geometries.

Calculations for planning the experiment in the TOKE project have been performed with the FLUENT CFD code. Thermal stratification in the T-joint simulating the connection between the hot and the cold legs of the primary circuit has been calculated by using the two-layer zonal model for turbulence.

A method is being developed to transfer data from CFD calculations to the structural analysis codes. Some preliminary capabilities are already in test use. Preliminary structural analysis was carried out using the surface mesh of the CFD model. Temperatures calculated with FLUENT were transferred to the input file needed for the ABAQUS code. These numerical results will be used for further planning of the tests.

Numerical 2D and 3D finite element crack propagation simulations have been performed with the Gurson model (Talja 2000) for three-point bending specimens with varying dimensions (Laukkanen 2000). The applicability of different constraint parameters is being evaluated and compared on the basis of the results. Evaluations concerning the measuring capacity of small specimens for ductile fracture resistance determination are being carried out.

Transferring tool for detonation loading transients was developed in co-operation with VTT Energy. This tool was used in transferring the hydrogen detonation load transient to the structural analysis model of a reinforced concrete wall. The transfer program for importing the pressure transients calculated by the DETO code to the structural analysis input file has been tested and taken into use.

The load carrying capacity of a reinforced concrete wall was studied. First, linear analyses were carried out and the load carrying capacity was evaluated based on codes and standards. Materially non-linear dynamic structural analyses were carried out. According to these analyses the wall seems to resist quite well the pressure increase before the peak detonation. The duration of a detonation is short compared with eigenperiods of the structure. The wall may somehow survive a detonation peak. But, the relatively slowly decreasing static type over pressure after a detonation peak severely damages the wall. Also, the decrease of the constant over pressure after the detonation due to the ventilation was assessed. The effect of this pressure decrease on the structural integrity of the reinforced concrete wall was predicted.

The post-processor program WSTRESS developed at the University of Illinois for the micromechanical analysis of brittle crack growth risk has been taken into use and it will be tested within the ongoing phase of the ESIS TG8 round robin.

A contribution to the final evaluation of NESC1 has been made. For NESC2 additional material testing has been performed using both deep and shallow cracked specimens. Fracture toughness behaviour of the material followed the Master Curve.

Main objectives in 2001–2002

The target is that quantitative fracture toughness estimates can be derived reliably from small material volumes and from minimum information of the material properties. Also, quantitative crack arrest toughness estimates can be derived from toughness properties of brittle fracture initiation. Capability to numerically simulate structures realistically under thermal and impact type loading conditions will be developed and verified.

The results by improved micromechanical fracture models is better understood. Considerably improved accuracy and reliability of safety assessment is obtained through evaluations, and more reliable computer codes are available through verification of codes in international benchmarks and networks.

Applications

The main application is the assessment of the ageing of equipment and structures. The targets are heavy steel and piping structures in power plants as well as reinforced concrete structures like containments, floors and walls. The accurate and reliable determination of material properties creates a good basis for structural analyses. The results may be applied in the analysis of structures affected by pipe whips and water jets due to pipe breaks, explosions and drop of a heavy specimen. The results also facilitate analysis of structures under complicated fluid dynamic conditions. A test will be carried out in order to verify numerical results of a structure under impact type loading.

Note: Special reports on the STIN project in chapters 5.2 and 5.3.

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2.3 In-service inspections and monitoring (INSMO)

During the in-service inspection the non-destructive testing (NDT) methods are applied to examine the structural integrity of the critical components. Therefore the reliability of techniques and procedures applied is of vital importance. Systematic qualification methods are seen as a very important approach to measure and prove the real effectiveness of the inspection systems. At the moment the qualification is also considered as a very effective way to enhance the reliability of the whole inspection system that includes following three main components: equipment, procedure and personnel. The main tasks of the project are linked to the establishment and development of the Finnish qualification system, where the guidelines designed in European co-operation are followed. Material monitoring techniques and also inspection methods of reinforced concrete structures are considered as special issues.

Main objectives in 1999–2000

Ultrasonic testing capabilities are enhanced by developing testing practices of mechanised inspections and by implementing a computer program for modelling of ultrasonic testing. The Finnish NDT qualification process is supported through a literature study about the essential inspection parameters applied in the technical justification. Development and verification of artificial test defects to be used in practical test samples was seen as an important part of the qualification development work and the research on them was started by manufacturing and testing of the first set of ultrasonic reflectors. Through participation in the AMES-NDT network (EU network for aged materials evaluation studies) relevant information is received about the NDT techniques to measure material properties. In the inspection of the reinforced concrete structures it was seen necessary to make a survey about the relevant components and construction details that could require new inspection or monitoring applications.

Main results in 1999–2000

The qualification of NDT methods used during in-service inspections of the nuclear power plants (NPP) was considered in two literature studies made to support the application and development of the Finnish qualification scheme. The new procedure requirements set by validation and mechanisation have been considered in the guideline paper, and a literary study has been made covering the use of parameters to be analysed in the technical justification during the qualification process. These two studies have been merged to one report that is published by STUK (Sarkimo 2000).

The creation of **ultrasonic simulation and modelling capabilities** has been started by purchase and installation of two computer programs applicable for different purposes.

One of these programs is capable to create three dimensional simulation models of ultrasonic inspection cases. 3D models of the components to be inspected can be created and imported to ultrasonic inspection modelling program. The probe movements on the component can be made "manually" or using a virtual scanner. An example of mechanical scanning of a simple pipe component is given in the figure 4. The simulation shows the path along which the probe is moved and reflections of the ultrasonic beam in the component to be inspected, as well as the material volume covered by the scanning. The other program is able to create two dimensional simulation of the ultrasonic wave front propagating in material. Simple reflectors can be introduced in the material to study behaviour of the wave front at a reflector.

The computer simulation and modelling of ultrasonic inspection is seen as a very important tool when designing an inspection. In many cases the inspection can be designed more efficiently if one has a chance to simulate the inspection scheme before the details of the procedure are decided. This approach is particularly useful when application is complicated or accessibility of the component is limited and coverage and performance of the inspection shall be carefully checked. Also the schedule of the inspections is tight during the outages of NPPs and it is necessary to be fully prepared before taking the equipment to the component. These actions can be greatly assisted by computer simulations at significantly reduced cost. The propagation of the ultra sound in the component and reflections at different defects can be studied to predict the performance of the inspection. The propagation of the sound beam and its reflections from the straight and curved surfaces of an artificial defect can be seen in the figure 5.

The **qualification** of NDT methods requires in many cases practical assessment of the capability of the inspection system. This can be carried out by a test where a test block resembling the real structure is inspected using the inspection procedure. Real, realistic or artificial defects shall be included in the structure in order to be able to assess the detectability of the defects when the inspection system is applied. In the case of ultrasonic testing different types of reflectors are introduced to represent cracks. A basic set of ultrasonic reflectors, applicable in validation trials, has been produced using electro-discharge machining (EDM). Ultrasonic signals from the reflectors have been recorded, analysed and reported. Also hot cracks produced by special welding technique and some fatigue cracks are available for ultrasonic measurements. The assessment of applicability of different types of EDM notches is now continuing. At the moment the Finnish qualification system is searching its forms and the practical methods of implementation are under development. During 1999–2000 the Finnish qualification organisations have processed the first practical cases (Paussu et al. 2000). Thus, all achievements supporting qualification development process are useful and can be utilised soon.

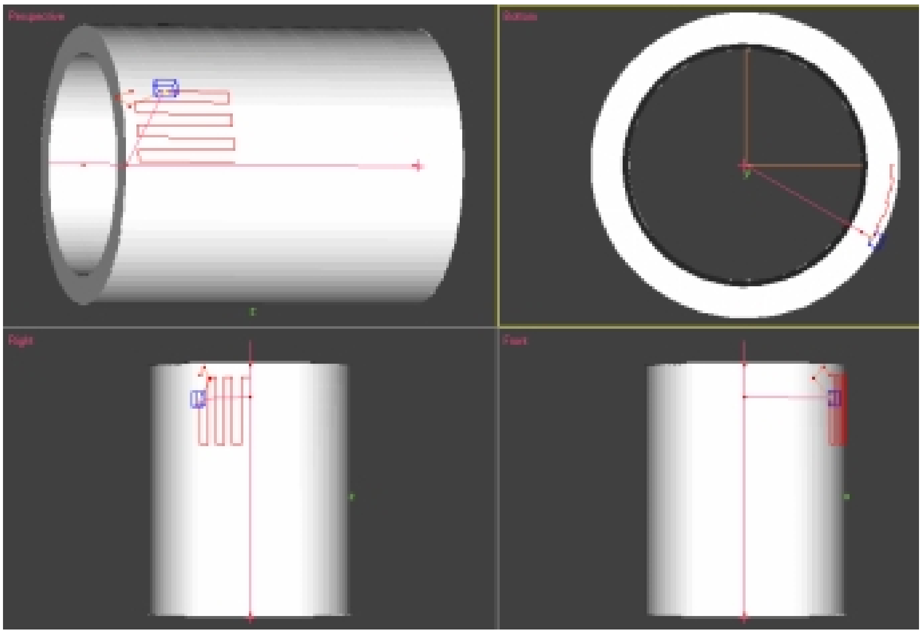

Figure 4. Simulation of mechanical ultrasonic inspection of a pipe section.

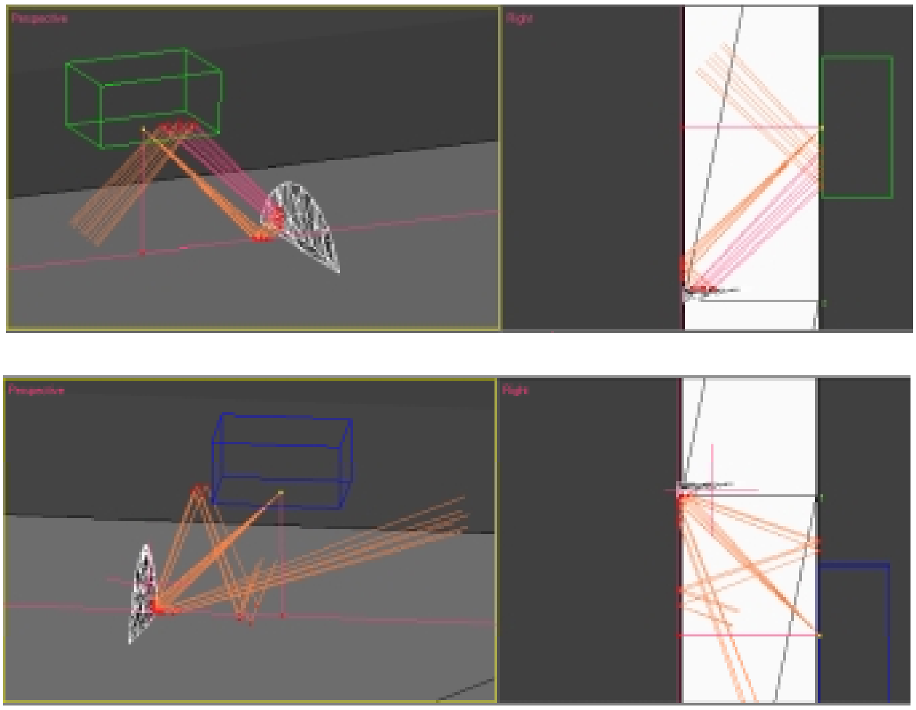

Figure 5. The two figures on the top show the normal reflection of the ultra sound beam at the straight surface of an artificial reflector. The sound is returning to the probe indicating the existence of the reflector. The two figures at the bottom show the splitting of the sound beam when it hits the curved and skew surface of the reflector. No portion of the reflected sound is hitting the probe and the reflector is not indicated.

Material monitoring techniques and their development have been the objectives of the participation in the work of the European AMES-network. A round robin project (GRETE) has been prepared by the AMES/NDT group to be started in the EU fifth framework programme. Measurements applying ultrasonic method are performed during this year to assess the capability of an in-house developed transducer. Material used for NPP valves is available in different conditions and the possibilities to measure the ageing of the material can thus be assessed.

The **reinforced concrete structures** and other critical construction details and their degradation mechanisms of NPP buildings have been identified in co-operation with representatives of the utilities and Finnish regulatory body. During the several visits to the NPPs these structural details have been examined in practice to find out applicable measurement techniques for condition monitoring. Also some preliminary trials using some measurement methods have been performed. A confidential status report has been complied, where a review about the damages of the concrete structures and the methods for monitoring and inspecting them is included. This summarises also the critical structure details in the Finnish NPPs considered during the first year of the project as well as the measurement trials performed.

Main objectives in 2001–2002

The modelling of ultrasonic testing will be developed by applying the available programs to real and test structures. The results of the test component simulations will be compared with the actual inspection results which will produce validation cases of the programs available. The applicability of the new tools in the market will be assessed and possibilities to purchase more accurate modelling programs will be considered.

The production techniques and validation of the test defects in the qualification samples are considered as a very important area. The production and tests on different types of ultrasonic reflectors will be continued. The ultrasonic response from the test reflectors will be assessed to justify their applicability in different simulation tasks. Also attempts to purchase real defects will be made. These real defects would be useful as references to the artificial defects. During the last phase of the project at least part of the test samples will be tested by destructive methods to verify the final condition and dimensions of the defects and reflectors applied in the project.

The material condition monitoring will be studied and developed using ultrasonic method. The VTT material probe will be applied to different measurements. The material conditions change caused by different ageing effects is seen as the most important application area.

Measurement trials will be performed on concrete constructions and other structural components of NPP buildings to develop applications for their condition monitoring. Ultrasonic measurements will be applied to monitor condition and possible cracks of the concrete structures. High energy X-ray unit will be applied to locate the position and lay-out of reinforced steels in thick concrete walls.

Applications

When NDT methods are applied every new system and procedure must be carefully tested and evaluated. The performance should be shown in a systematic and traceable way and necessary documentation compiled. Especially when inspection of critical components is considered the requirement for reliable assessments of NDT systems must be fulfilled using an approach that can be approved by the Finnish regulatory body. The gained know-how about the organisation and documentation of qualifications as well as about qualification sample production can be applied also in other industrial sectors and in international projects.

Ultrasonic inspection of a complicated component is a difficult task requiring careful design. The performance and its limits can be quickly analysed in some extent by simulations. Valuable information can be gained in the design phase and also supporting evidence material can be produced for qualification. Such an approach can reduce considerably the time and costs needed in the application design and verification phase.

Note: A special report on the INSMO project in chapter 5.4.

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3. Accidents: Main goals and results

3.1 Topics in transient behaviour of high burnup fuel (KOTO)

As soon as the use of nuclear fuel in the core of a reactor begins, a long and versatile chain of events is initiated that changes the fuel in far-going ways. After these gradual changes, the properties of the parts and materials of a nuclear fuel rod are generally far from those they once were with fresh fuel.

With accumulating experience, with developing material sciences, and with improved calculation tools and hardware, the research and industry have been able to push the fuel discharge burnups to levels that are tens of percent higher than before. There is a clear economic incentive to continue. For instance, a rough estimate gives a total annual saving of € 4 million in the Finnish plants, if a near-term goal of burnup increase of 5 MWd/kgU could be licensed. Of this about 60% is a direct saving in fuel purchase, 40% in spent fuel handling and disposal cost. The upper limit for the average discharge burnup of any single assembly is currently set at 40 MWd/kgU by the regulator.

From steady state testing and by irradiation experience, it has been quite well demonstrated that the reliability and safety of operating fuel, the natural prerequisites of preserving the gained savings, are maintained beyond the current burnup limits in normal operation. The high burnup fuel properties, that sharply deviate from those of the fresh, however, have potentially much stronger influence in transient and accident situations. The data bases behind the criteria extend to the burnup regimes now applied only by their small parts. Unfortunately, becoming convinced about the proper performance of high burnup fuel in transients and accidents is not straightforward.

There now seems to be, after the infamous research results – and extensive analyses that partly relieve the most urgent concern – a consensus that fuel accident behaviour deserves a systematic re-evaluation. A number of international efforts has been newly launched to this end, most notably the experimental OECD – IPSN CABRI Water Loop Project to study reactivity initiated accidents (RIA).

The demand of improving the situation is reflected in the objectives of the current FINNUS/KOTO project, which is the follow-up of a succession of tasks. Apart from that, the utilities Fortum and TVO have recently taken a lead to finance the participation in the CABRI Water Loop Project, with partial support from the state through National Technology Agency Tekes, and with VTT Energy as the main Finnish contractor. The FINNUS/KOTO project continues to host several other studies on high burnup fuel performance.

Physical background

The changes with burnup in fuel materials come from thermal, chemical, crystallographic, irradiation damage and, in the pellet, neutronic contributors. The cladding will oxidise, it will harden from irradiation and it will pick up hydrogen, which originate from the coolant after radiolysis and consumption of the oxygen from a growing oxide layer. The absorbed hydrogen form hydride precipitates in the metal that effectively make the cladding become brittle. The oxide layer will eventually grow also on the inner surface of the cladding by a reaction with UO2; occasionally with an actual chemical bonding of the two. In the pellet, the irradiation and temperature cause restructuring of the ceramics, solid and gaseous fission products cause swelling, and chemical and structural changes give rise to reduced thermal properties.

The most specific high burnup phenomenon is the so-called rim that will form on the pellet. In a thin layer on the pellet, the original microstructure is totally lost, the porosity and plutonium content and thus burnup and power density are high, and small grains are prone to separate. The releasing and released fission gases cause additional swelling and finally increase the pressure inside the rod. Caesium, iodine and other volatile and aggressive fission products gather in cooler spots, often next to pellet interfaces and may enhance stress corrosion cracking.

The accident scenarios most prominently under consideration are the more or less conventional RIA (reactivity initiated accident) and LOCA (loss-of-coolant accident; large break). In RIA, increase of power and temperature are violent and these concentrate on the mechanically weak pellet rim area. In LOCA, in turn, there will always be a high potential of large deformations in the cladding, as the cooling as well as the system pressure will be quickly lost. Whenever the cladding is fractured in an accident, there will be a concern of fuel relocation and dispersal from the damage and potential interaction with the coolant. Example of the thermal behaviour of a rod in RIA is seen in figure 6.

Regulatory background

STUK has issued a new set of regulatory guides that define the requirements on fuel performance, mainly in (STUK 1999). In these guides, directions have been adopted that are in concert with the ideas that an OECD body has been recently formulating (CSNI 1999). This is evidently the first occasion in the world that they are coming into

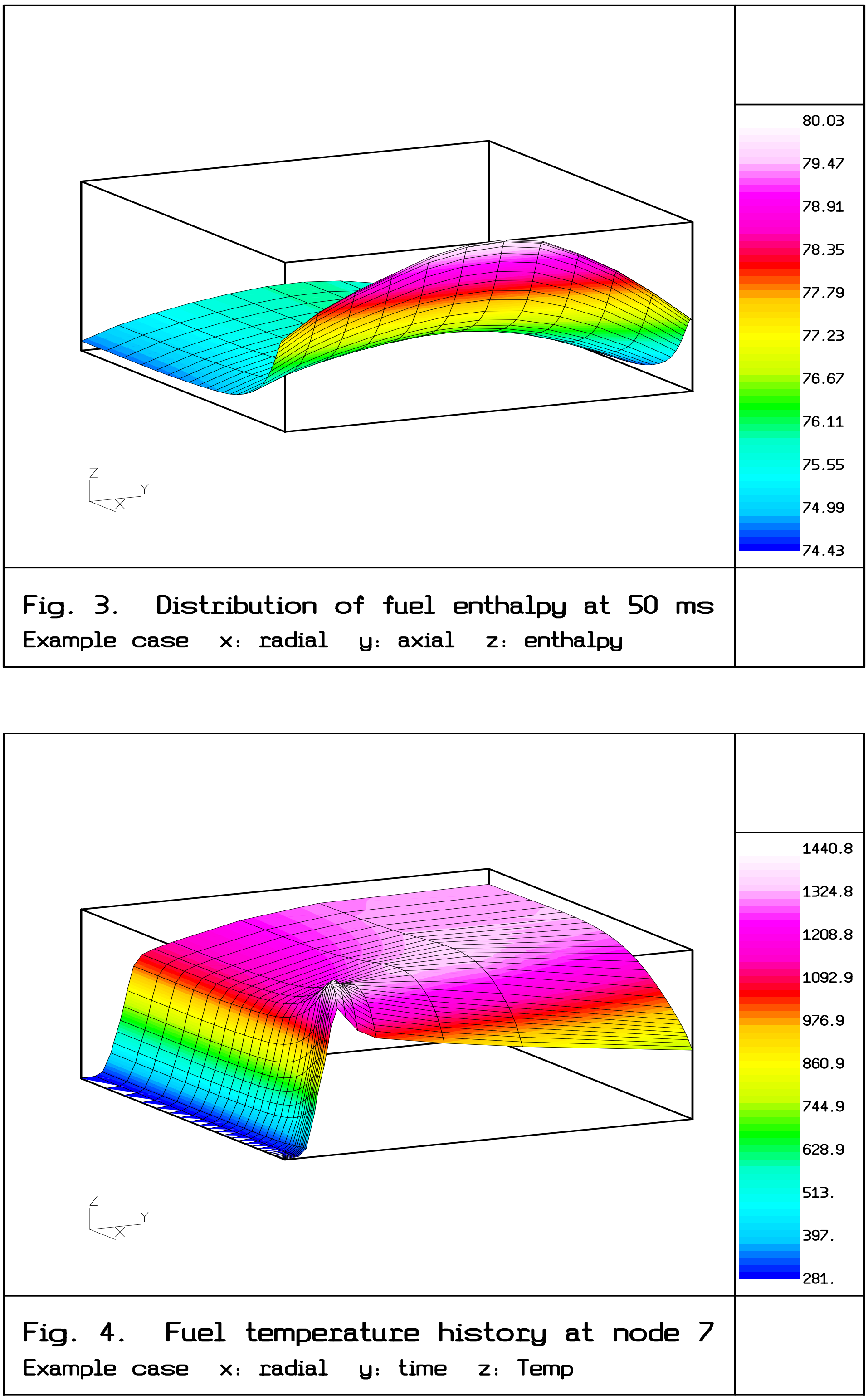


Figure 6. Distribution of enthalpy and evolution of temperaturte distribution for a generic fuel type in a reactivity initiated accident (RIA).

effect. In these, the traditional categories of design basis accidents have been abandoned and a wider spectrum of events is introduced, with the probability of an event given the more restriction the more severe its consequences are. This is physically sound but will place high demand on the theoretical and analytical comprehension of the scenarios. The final goal for calculations is generally a probability distribution of rod failure with certain feasible uncertainties of the initial and boundary values. This would require using probabilistic methods, development of appropriate failure criteria, and proper knowledge of material properties from representative conditions.

The above requirements of the reactor and fuel types in use, pose a vast field of new development and applications. The VTT Energy projects together will work to gradually meet this goal. Some of the current tasks are mentioned in the following.

Results of the FINNUS/KOTO project in 1999–2000

The fuel performance codes are roughly categorised in steady state and accident behaviour applications. Normally, the former serve to simulate the long-term power reactor operation and to initialise an accident code. The neutronic and thermal-hydraulic boundary conditions come from specialised system codes.

The steady state code of which VTT has the longest exerience is called ENIGMA, originally acquired from that-time British Nuclear plc. The code has been adapted to VVER and BWR conditions with subroutines developed to include VVER specific material properties. A special probabilistic version has been created and applied by VTT to obtain statistical distributions of parameters starting from assumed distributions of the input data, uncertainties of models, and distributions of operating conditions. (Ranta-Puska et al. 2000). A less common application has been studies of some oscillations, which has been possible thanks to the non-stationary heat transfer description included (Ranta-Puska 1999a). In addition, a link deck was created that will allow an accident code to be initialised with the validated ENIGMA code.

There are constant efforts on steady-state validation, separate effect simulation, and material property elaboration (Ranta-Puska 1999b) on the basis of collaboration with the OECD Halden Reactor Project. VTT tools have been applied on extensive analyses of the VVER fuel irradiations performed in Halden over past several years.

In collaboration with the French safety research organisation IPSN, VTT has installed and used the IPSN SCANAIR code, developed to simulate a fuel rod in RIA mainly (Federici 2000). Recently, VTT Energy has taken an active role in the further development of SCANAIR. In a master's thesis work, a model for creep, an essential phenomenon in high-temperature transients, was incorporated in SCANAIR (Knuutila 2000). The work includes a literature survey on creep models.

The USNRC has launched programmes to renovate their steady state and accident codes. These efforts partly continue, as it appeared that satisfactory results could not be reached without profound re-writing of the coding. Necessary high burnup descriptions were included. The Finnish STUK has started a close collaboration with the NRC, with active participation from VTT to the development of the two NRC codes, viz. FRAPCON-3 for steady state and FRAPTRAN for accident conditions. The competencies on fuel behaviour and thermal hydraulics have been combined to introduce, in collaboration with the FINNUS/READY project, an interactive thermal hydraulics model in FRAPTRAN that is based on VTT in-house GENFLO formulation, with expected completion in 2000. With the parallel installation and collaborative development of the NRC codes, STUK wishes to maintain independence of the utility analyses and a forefront position in the international development in this sector as well.

Main objectives of the FINNUS/KOTO project in 2001–2002

Within the FINNUS Programme, selected studies will continue to improve the understanding of phenomena and to promote the performance of analytical tools.

The further development of the ENIGMA steady-state code is focusing in improving the descriptions of high burnup phenomena, fission gas release in particular. The USNRC codes with upgraded thermal hydraulic subroutines will enter into systematic testing and application. The new high burnup material data will be incorporated into the performance models. Flexibility of the codes and features that will make them easier to use will be added, together with elaborated post-processing capabilities.

Code validation, qualification of test results, and analysing separate effect tests will be actively continued under the collaboration with the Halden Project collaboration.

Memberships in bodies of international organisations continue. The IAEA is expected to launch another co-ordinated research programme on fuel performance code comparison. Post-graduate studies and other educational efforts will be encouraged.

The Finnish CABRI project

Finnish participation in international experimental projects now continues outside the FINNUS Programme, but in close co-ordination with its fuel studies. Through a VTT-utility consortium, preparations are well under way to participate in the OECD-IPSN Water Loop Project 2000–2007, an experimental research programme on high burnup

-light water reactor fuel in prototypic RIA conditions. The key experiments that the high burnup issue is resting on were once made by IPSN in CABRI reactor in CEN Cadarache, in a test rig with sodium coolant. An internationally sponsored € 50 million project has now started to build a water loop in the CABRI reactor with planned 12 test altogether of which ten in the new facility. With water as the coolant, one aims at a proper simulation of its interaction with fuel beyond the first fractions of a second. The probable scenario depends on interactive factors in the manner schematised in figure 7. In the planned tests in the pressurised water loop, the project will address the interactions from the onset of the departure from nucleate boiling (DNB) up to the behaviour of dispersed fuel in steam. Rod burnups of up to 60-80 MWd/kgU and representative cladding materials are anticipated. The schedule of the tests and related Finnish activities are planned as in table 2. The project will have a close connection with the similar research at the NSRR laboratory of JAERI of Japan.

Attached with the CABRI Project, the French separate effect programmes PROMETRA and PATRICIA for material testing and thermal hydraulics, respectively, will become within reach of the participants. In the former, the cladding of the M5 type, similar to the VVER cladding, and high temperature ranges of Zircaloy will be addressed. In PATRICIA, transient heat transfer coefficients will be determined.

The USNRC has started an extensive out-of-pile support programme for high burnup cladding behaviour, to be performed at the Argonne National Laboratory. Irradiated claddings from actual PWR and BWR rod will be tested in various types of experiments. The test types include oxidation tests in steam, simulated LOCA criteria tests with internal pressurisation and simulated pellets, plus ring and axial tensile test, and biaxial burst test, with RIA and LOCA-related conditions separately. The total number of tests with irradiated samples will approach 350. The cladding samples will be from rods with burnups of 50 MWd/kgU, typically. Zr-2 liner, Zr-4, and Zirlo type materials are included. The results are expected to become shared by the CABRI Project participants.

Conclusions

In reactor technology and safety, fuel performance is in a special position, as fuel is the part of the system that will face constant changes in design and operational data. The variety of reactor types and materials, the difficulty and high expense of performing the experiments, and the difficulty to theoretically and analytically cover the many affecting factors, together with the requirements of the new regulatory practices, place a great challenge to high burnup fuel studies, however. It is acknowledged that a small part of the whole can be addressed at a time. On the other hand, the Finnish participation and

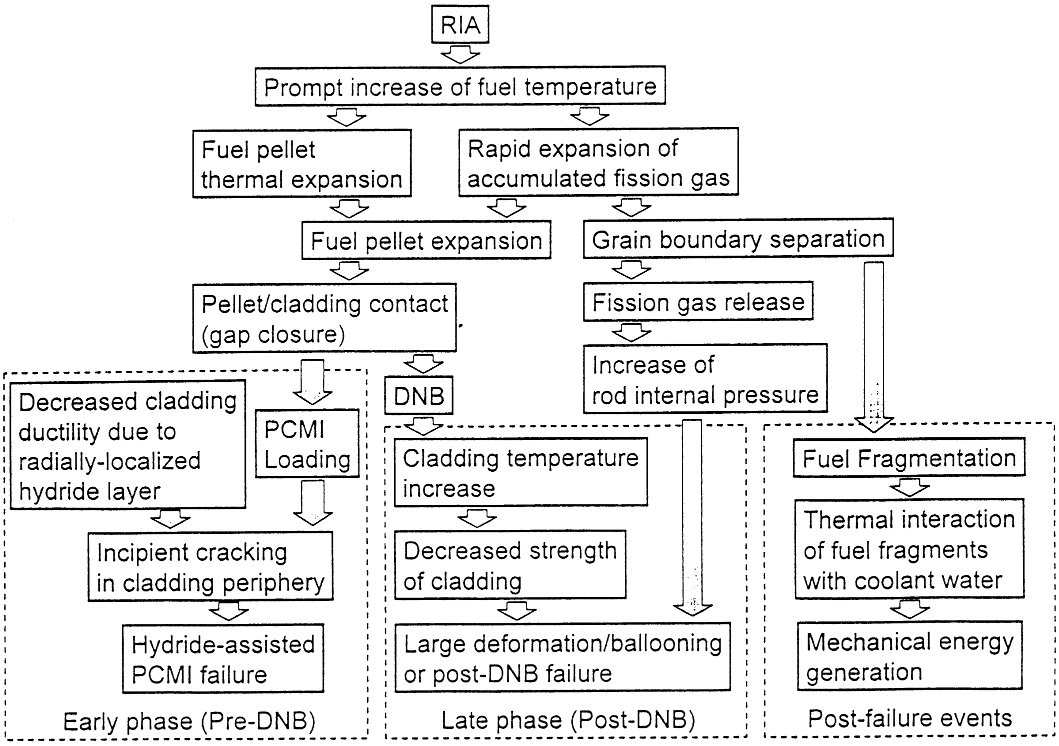
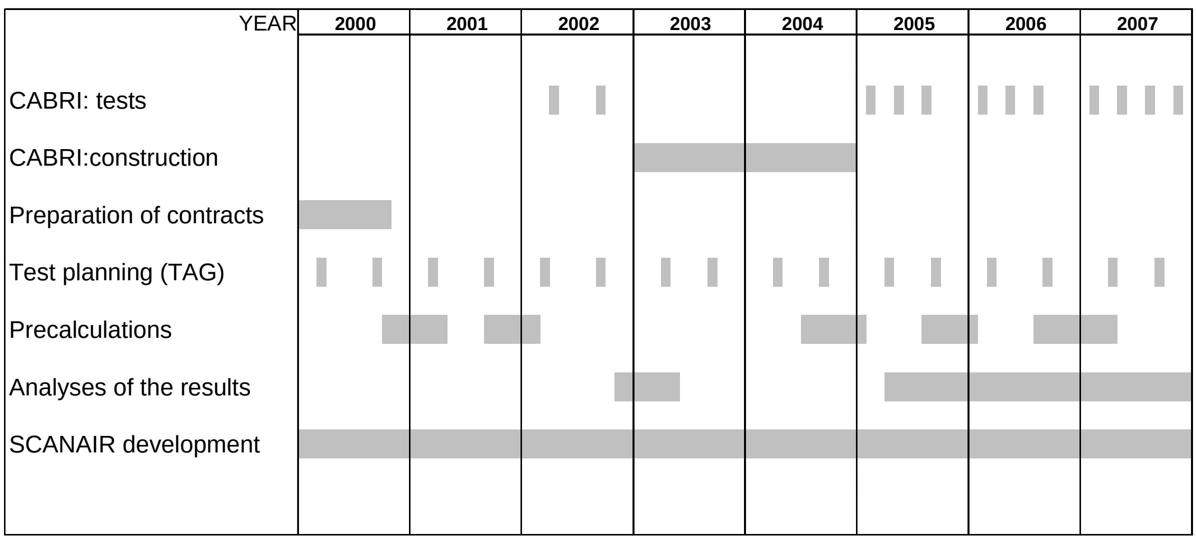


Figure 7. Relations of LWR fuel rod behaviour phenomena and scenarios in RIA transient with water coolant.

Table 2. Improving fuel efficiency by increasing burnup (CABRI). Time table of CABRI tests, planning, and Finnish studies in 2000 through 2007.



presence in the essential international research efforts are well advanced, and active and competent participation in the actual research and development work is on-going and anticipated to increase. The authority-sponsored parallel development and application route based on the USNRC codes and collaboration will accentuate the role of VTT as an independent consultant. Also, encouraging educational progress has been made after a long period of alarmingly low activity in this respect.

Note: A special report on the KOTO project in chapter 5.5.

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3.2 Reactor physics and dynamics (READY)

VTT Energy has created and intends to maintain a computer code system and competence for carrying out all reactor physics calculations needed in Finland, as well as a comprehensive and independent computer code system and expertise for reactor safety analyses, providing tools from basic nuclear data to three dimensional transient and accident analyses.

Main objectives in 1999–2000

In reactor physics the main objective is to ensure reliable and continuous use of the wide range code system both by updating and validating the codes and the ways to use them, and training of new personnel. In 1999–2000 special emphasis is in learning to use Monte Carlo Technique, applying the nuclear data processing system NJOY'97 and validating three-dimensional out-of-core calculation systems against experimental benchmarks. In order to be able to model cores with increased heterogeneity, an advanced nodal method (including pin power reconstruction) is to be developed.

In reactor dynamics the objectives are to complement and validate the calculation system for complex scenarios, such as ATWS, boron dilution and BWR core stability. The plan is to validate the three-dimensional reactor dynamics codes TRAB-3D and HEXTRAN against plant measurements and international benchmarks. Co-operation with the KOTO project includes improving of the fuel models of the reactor dynamics codes and introducing thermal hydraulic models from the existing codes to the fuel behaviour analysis codes. The thermal hydraulic models of the dynamics codes are improved by taking into use the numerically accurate solution method PLIM. The CFDPLIM solver itself is also further developed to have a new more robust version for the coupled calculations. To model complicated accidents, correlations for counter-current flow are studied. Possibilities of the six-equation model SFAV in modelling thermal hydraulic phenomena is studied.

Main results in 1999–2002

Validation of three-dimensional reactor dynamics codes

The three-dimensional reactor dynamics code TRAB-3D (Kaloinen & Kyrki-Rajamäki 1997) developed at VTT has been validated for coupled neutronics and thermal-hydraulics analyses of PWRs and BWRs in square lattice geometry (Daavittila et al. 2000a, Zimin et al. 2000). In addition to the core dynamics TRAB-3D includes BWR circuit models; the combination has been succesfully validated by calculating real TVO plant transients (Daavittila et al. 2000b). TRAB-3D has also been validated by calculation of the OECD/NEA Main Steam Line Break benchmark (Ivanov et al. 1999). All the three phases of the benchmark were calculated duly and with good results using TRAB-3D coupled with the SMABRE circuit model of the PWR (TMI-I). (See also chapter 5.7.)

Improved neutronics modelling

Static and dynamic analyses of reactors are normally carried out using nodal methods, in which the fuel assemblies are modelled with axially divided homogeneous volumes. A new three-dimensional neutronics nodal model based on the analytical function expansion method (AFEN) (Noh & Cho 1994) has been developed at VTT. The AFEN & Cho model is needed in modelling of new complicated fuel types. In addition to improved accuracy within the reactor core, use of analytic expansion functions enable calculation of the neutron flux in reflector areas, which could eg. improve the accuracy of the ex-core detector flux estimates. Moreover, the pin power reconstruction model (Mattila 1999a, Mattila 1999b) developed earlier in the project is directly applicable within AFEN. The model has been succesfully tested against IAEA-2D benchmark (Argonne Natinal Laboratory 1977). (See chapter 5.6.)

HEXTRAN is a three-dimensional hexagonal reactor dynamics code developed at VTT (Kyrki-Rajamäki 1995). A model of the out-of-core detector signals with precalculated kernels was implemented in the HEXTRAN three-dimensional reactor dynamics code for VVERs utilizing Monte Carlo calculations (Kaloinen et al. 1999, Wasastjerna & Siltanen 2000). The comparisons against Loviisa and Czech Mochovce rod drop experiments showed increased consistency (20→10% difference), but still more advanced neutronics modelling is needed to achieve full agreement.

Modelling of new fuel types

New fuel designs and intentions to increase the burnup of the nuclear fuel sets new requirements for calculation models used in safety analyses. The problem has been approached from two sides: by improving the fuel models of the dynamics codes and by introducing advanced hydraulic models in fuel codes.

New models of the radial heat generation within the fuel pellet and of the gas gap behaviour have been implemented in the reactor dynamics code TRAB (Syrjälahti 2000). In the new models the heat generation depends on radial position and burnup; and the gas gap conductance depends on the temperature, pressure and width of the gas gap as well as on the free volume of the fuel rod. In addition, the effect of the possible contact between the fuel pellet and cladding on conductance has been included in the model. New models have been tested in a control rod ejection case.

FRAPTRAN is a computer code used for transient and design basis accident analysis of the behavior of a single fuel rod under off-normal reactor operating conditions (Siefken et al. 1981). The fuel behavior model of FRAPTRAN has recently been upgraded but its hydraulic model is still based on the old RELAP homogeneous model. At VTT FRAPTRAN has been coupled with a more sophisticated hydraulic code called GENFLO (GENeral FLOw) that has been developed at VTT. The hydraulic models and calculation routines of GENFLO are similar to those of the SMABRE (Miettinen & Hämäläinen 2000) and RECRIT (Miettinen et al. 2000) codes. Hence, GENFLO is based on fast, non-iterative five-equation model including also rewetting capabilities up to the melting temperatures of the fuel rods. The first version of the new coupled code is being tested in co-operation with the KOTO project, see chapter 3.1.

Advanced thermal hydraulics modelling

Piecewise Linear Interpolation Method (PLIM) (Rajamäki & Saarinen 1991) is a hydraulic solution method, developed at VTT, that is aimed at improving the accuracy of the reactor dynamics codes in challenging flow conditions. The PLIM method eliminates numerical diffusion and dispersion, which improves eg. tracking of boron and temperature fronts in transients. To improve the capability of PLIM in the cases where the derivative terms are less important than the source and coupling terms, a new numerical iteration scheme has been developed. The scheme, that resembles a normal solution method of conventionally discretized equations, was found necessary to treat the thermal hydraulics equations of reactor dynamics codes.

The pump model of the 1-D TRAB has been implemented in the version with the new hydraulics model PLIM, and it has been successfully tested with several transients. The HEXTRAN-PLIM core model has been shown to give consistent results with the old version in mild transients. Various drift-flux correlations were programmed and tested in complicated flow regimes such as counter-current flow and reversed flow. Good candidates were found for further trial studies.

Separation of the Flow According to Velocity (SFAV) (Narumo 1997) is a two-phase flow formalism developed at VTT. The amplitude growth of the propagating voidage waves, earlier calculated at 7 MPa, were recalculated with SFAV at 0.1 MPa in order to obtain information for experimental work concerning significance of the phase separation phenomena.

Updating and validating the reactor physics code system

In reactor physics the Monte Carlo method is chosen to solve complex problems in criticality and radiation shielding. Several advanced features of the MCNP Monte Carlo code (Briesmeister 1997), such as the differential operator perturbation technique to evaluate small reactivity changes or variance reduction techniques in deep penetration shielding and streaming problems, have been validated and taken into use. The Monte Carlo method has also been applied to study the VVER-440 control element efficiency and the Tokaimura criticality accident in 1999 (Tanskanen 1999a).

The NJOY nuclear data processing system is used for producing pointwise and multigroup nuclear cross sections and related quantities from evaluated nuclear data in the ENDF format (MacFarlane & Muir 1994). NJOY was applied to generate multi-temperature neutron cross section and thermal scattering data for the MCNP Monte Carlo particle transport code from the ENDF/B-VI revision 5 evaluated nuclear data (Tanskanen 2000a). The new data enables wider use of accurate Monte Carlo analyses.

An international benchmark, launched by the OECD/NEA, to examine the current computation techniques used for calculating neutron and gamma doses to reactor components, revealed that three-dimensional neutron fluence calculations provide results that are significantly more accurate than those obtained from two-dimensional calculations (OECD/NEA Nuclear Science Committee 1999). Previously, the two-dimensional DORT discrete ordinate transport code has been used at VTT for out-of-core flux calculation (Wasastjerna 1992). Now, the three-dimensional TORT discrete ordinate transport code has been validated against the VENUS-3 benchmark (Leenders 1988). A discrete mesh generator program DIMER (Tanskanen 2000b) has been developed at VTT to automate mesh, material distribution and source distribution generation for TORT calculations. The calculated benchmark results were in good agreement with the reference results.

International research co-operation

International benchmarks have been utilised in validation and development of the reactor physics and dynamics codes. Not only has the accuracy of VTT's independent computer code system been demonstrated, but calculation of benchmarks has contributed to the international research contacts. VTT's expertise in three-dimensional dynamics calculations has been utilised in defining, solving and coordinating three-dimensional hexagonal dynamics benchmarks in the international co-operation on VVER reactor physics and safety (AER) (Kliem et al. 1999). Also criticality calculation codes have been validated within AER co-operation (Tanskanen 1999b). Moreover, VTT has succesfully participated in calculation of benchmarks launched by the OECD/NEA Nuclear Science Committee (NSC). Worldwide interest in coupled neutronics/thermal hydraulics calculations has also increased co-operation with the NEA Safety Committee (CSNI) (Kyrki-Rajamäki & Räty 1999, Kyrki-Rajamäki 2000, Mittag et al. 2000).

Education of experts

One of the goals of the READY project is to maintain the reactor physics expertise in Finland. The project supports both undergraduate and post-graduate studies. Two master's theses were completed in 1999 (Latokartano 1999, Mattila 1999a). The project employs presently seven young persons (YG): five post-graduate scientists and two research trainees. The YG has participated in international training courses and summer schools.

Main objectives in 2001–2002

In the near future the strategic objectives of the project will remain unchanged. In reactor physics the main objective is to ensure reliable and continuous use of the wide range reactor physics code system, that largely relies on internationally widespread computer codes acquired through the NEA Data Bank. Therefore, it is essential to maintain close contacts in the field of reactor physics with NEA Nuclear Science Committee. Especially participation in the work of the nuclear criticality safety groups is planned. Fruitful co-operation on VVER safety will also be continued within AER. Applications of the Monte Carlo method are extended. Further efforts in reactor physics are included in the development of the reactor dynamics codes.

In reactor dynamics the main objective is to converge all the accrued development into production versions of the codes. This includes development of the advanced neutronics (AFEN) and thermohydraulic (PLIM) models and their application to reactor dynamics codes; validation of the calculation system (NEA/NSC BWR turbine trip and AER asymmetric steam line break benchmarks); and updating the code manuals. Modelling of fuel transients is continued in close co-operation with the KOTO project and the NEA/CSNI working group on Fuel Safety Margins.

Applications

VTT's comprehensive and independent reactor physics and dynamics code system is applicable to safety analyses of BWRs and PWRs including VVERs. The code system has been widely used by the nuclear safety authorities and by the Finnish nuclear power companies as well as by customers abroad. The code system was extensively utilized in the analyses of the recent power uprating projects of Finnish NPPs: in the selection of new fuel types including criticality studies, and in the transient and accident analyses. VTT has for many years carried out the ICFM work for the Olkiluoto-1 BWR, utilizing also the expert system CORFU (Höglund & Latokartano 1999). The ICFM of the Loviisa VVERs is made with VTT’s HEXBU code (Kaloinen 1992) which was also used in the development of a new VVER-440 fuel design. Nuclear data for these analyses is generated by VTT with the CASMO code.

The power uprating, new advanced reactors and fuel types, new safety concerns, the trend towards higher enrichment and burnup as well as better optimized cores set new requirements for the calculation methods. More accurate modelling is needed to remove uncertainties, excess conservatism, unphysical fittings, and to analyse numerically challenging flow transients, where simplified assumptions can even lead to a false sequence of the events. Three-dimensional core dynamics also allows for modelling of mixed fuel loading without simplifying averaging procedures.

The expertise gained in the project has been in great demand. New improved reactor dynamics code versions have been applied to the safety analyses of the Finnish NPPs. The reactor physics expertise has been utilized in waste management, simulator applications, severe accident studies, materials research, fusion studies, as well as by the Ministries of Defence and Foreign Affairs. The expert knowledge has also be applied to international contract research, as well as to improve safety of VVERs in Central and Eastern Europe, eg. within Phare and Tacis projects.

Note: Special reports on the READY project in chapters 5.6 and 5.7.

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3.3 Thermal-hydraulic experiments and code validation (TOKE)

The objective of the thermal-hydraulic experiments and code validation project (TOKE) is to produce data for developing more accurate safety analysis tools and to enhance understanding of phenomena occurring during abnormal situations in nuclear reactors. The TOKE project is divided into three parts: 1) VVER related tests with the PACTEL test loop, 2) validation of thermal-hydraulic computer codes and 3) separate effect tests to determine boundary conditions for the assessment of the effects of a power plant lifetime. The VVER related tests focus on studying integral system behaviour during transients. The validation of thermal-hydraulic computer codes, especially the APROS system code, is closely connected to these tests. The separate effect tests produce data for computational fluid dynamics (CFD) calculations of loads caused by water flow and thermal stratification. These calculations are carried out in the structural integrity project (STIN) of the FINNUS programme.

Main objectives in 1999–2000

The main objectives of the VVER related tests in 1999–2000 were to examine a steam generator collector header rupture incident with the PACTEL test loop and to study the effects of non-condensable gases on heat transfer in a horizontal steam generator. In the code validation part of the project, the objectives in 1999–2000 were to validate the APROS code against OECD/NEA Main Steam Line Break benchmark (TMI-I) and against PACTEL results, to create a new APROS input model of the PACTEL test facility with the help of the GRADES user interface and to produce in-kind contribution for the CAMP agreement. The scope of the separate effect tests in 1999–2000 was to build a test-loop for carrying out experiments to study thermal stratification in a small diameter vertical tube. The goal was to provide boundary conditions for thermal fatigue analysis and verification data for CFD-codes. The codes are used in the STIN project to determine local loading effects due to water flow and thermal stratification. The results of the CFD-calculations are then transferred to structural analysis. Experiments focusing on water hammer phenomenon and experiments addressing a large break loss-of-coolant accident (LBLOCA) were also planned for 1999–2000, but have been delayed due to the reduction of resources, which are available in the TOKE project.

Main results in 1999–2000

Strong flow and core power oscillations were observed in **an ATWS test with core reactivity feed back** simulation (Riikonen & Kouhia 1999). The experiment simulated a control rod withdrawal transient when the core was operating with almost zero power. The operation of the pressurizer safety valves was simulated with two scaled down valves. As the main circulation pumps coasted down and the heat exchanger tubes of the steam generators were steadily uncovering, energy transport from the core to the steam generators decreased. The primary pressure increased and the pressurizer safety valves opened. When the core voided enough, the control value of power decreased below the maximum available core power. Then, the primary system began to oscillate. One cycle took 10–15 s. Due to the time constants of the power control system there was a 5–8 s delay before the measured core power reached the setpoint value of power. That is about half of the whole cycle. The core power was thus all the time in a phase where it amplified the primary system oscillation process. The oscillations continued to the end of the experiment. Because the maximum available core power was only 20 % of the scaled nominal power, the time scale of the phenomena in this kind of experiment is distorted. If the available power had been higher, the core would have voided earlier and the core power feedback could have functioned as anticipated in a real transient.

In the Loviisa nuclear power plant (NPP), the construction of the steam generator primary collectors will be changed. In future, if a collector rupture occurs, the leak flow area from the primary to the secondary side will reduce significantly. To investigate the effect of the leak restrictor on the system behaviour, two **collector header rupture experiments** were performed in the PACTEL facility (Riikonen 1999). The test configuration of the earlier steam generator tube rupture experiments was utilised (figure 8). The experiments were based on the current regulations for operator actions during a state of an emergency in the Loviisa NPP. In the larger break size experiment, the occurring phenomena were faster and the safety valve of the broken steam generator cycled few times and released secondary side inventory to the atmosphere (figure 9). The steam generator safety valve did not open when the restrictor (small break) was used.

The **effect of non-condensable gases on system thermal-hydraulics and on heat transfer** in a horizontal steam generator has been experimentally studied with single loop experiments in the PACTEL facility. Compressed air and helium acted as a non-condensable gas. Gases can have an effect on the system behaviour for example during severe accidents and in loss-of-coolant situations if accumulator gas (nitrogen) is released into the primary circuit pipework.

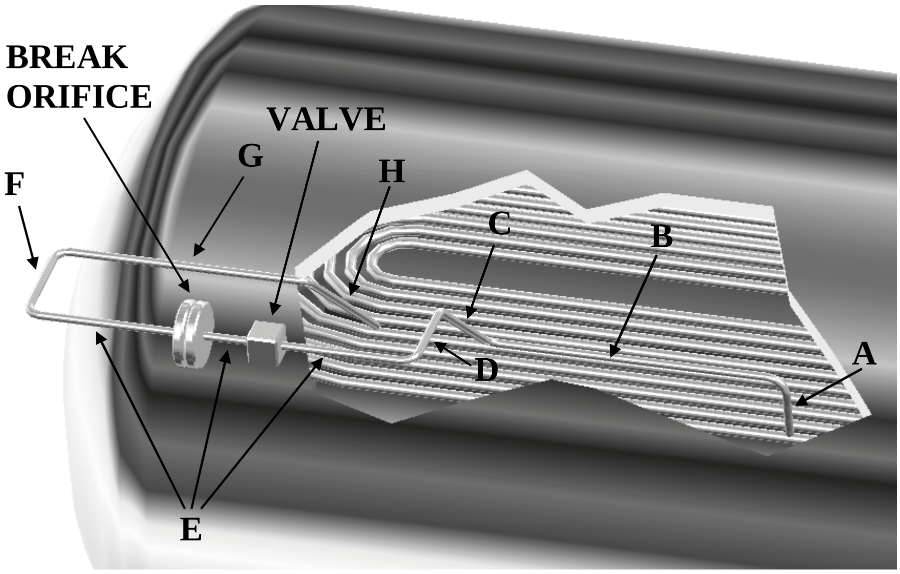

Figure 8. Break simulator in the steam generator collector rupture experiments.

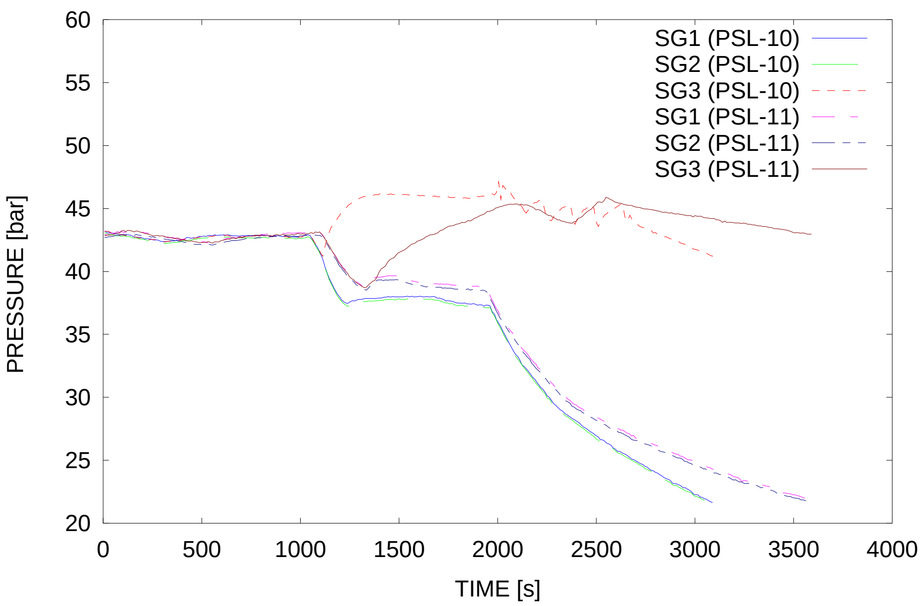

Figure 9. Secondary pressures with (PSL-11) and without (PSL-10) the flow restrictor in the steam generator collector header rupture experiments.

During two-phase natural circulation and boiler condenser modes the system response to the presence of non-condensable gases depends partly on the gas in question. If the gas is heavier than vapour (nitrogen and air for example), it may accumulate to the lowest tube rows of the horizontal steam generators while vapour continues to flow through the uppermost tubes. If the gas is lighter than vapour (hydrogen for example) the system behavior is expected to be different. The gas may lay in the top part of the steam generator tube bundle as vapour flows through the lowest tubes and condenses there. The internal circulation flow pattern of the steam generator tube bundle would therefore also be different in these two cases. When air was used as a non-condensable gas, the steam generator behaved as expected (figure 10). Air accumulated to the bottom tube rows and steam continued to flow through the top part of the bundle. However, the measured loop behaviour with helium as a non-condensable gas was much less different than expected from the case, where air was used. Helium did not accumulate clearly to the uppermost tubes. Instead, vapour and helium mixed with each other and the heat transfer from the primary to the secondary side deteriorated rather uniformly across the tube bundle.

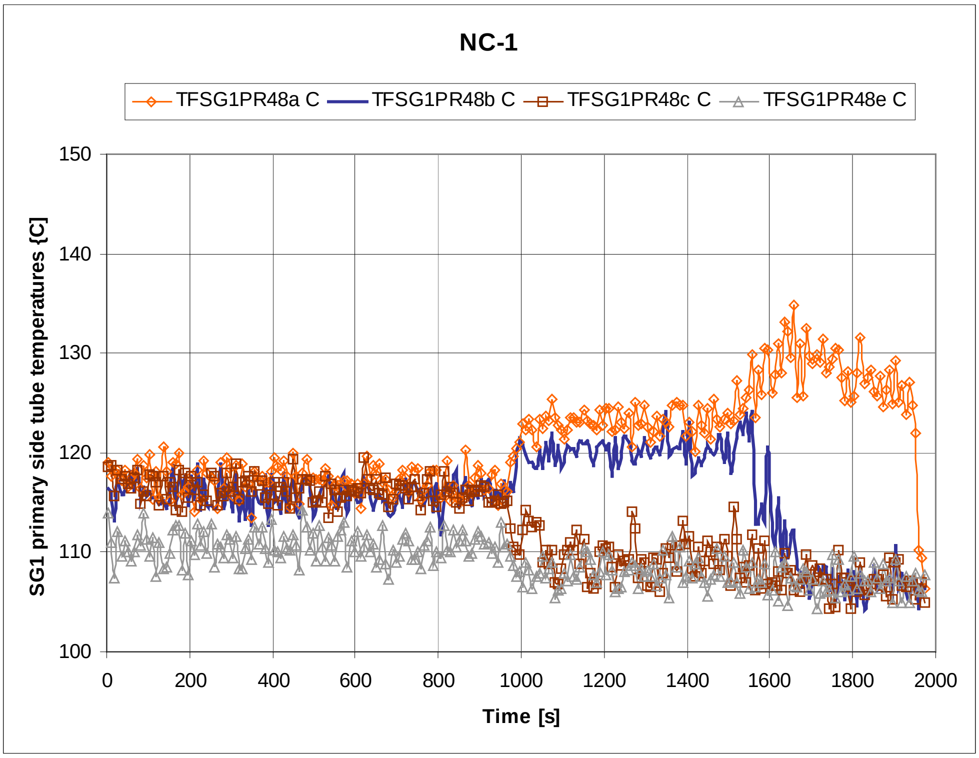

Figure 10. Heat transfer deterioration in the steam generator mid-elevation due to the injection of non-condensable gas as indicated by temperatures along a single tube (gas injections at 960 s and at 1500 s).

An extensive test series was carried out to **measure the pressure losses of the different PACTEL components** as a function of mass flow rate (Kouhia & Puustinen 1999). The pressure losses were determined both for normal and reverse flow direction. The three primary circuits, the pressure vessel and the pressurizer with the surge line were all measured separately. In addition, the pressure loss over a main circulation pump was measured. The highest local pressure losses, normalised against mass flow rate, were measured over the main circulation pump and in the pressurizer surge line.

In the **simulation of the OECD/NEA Main Steam Line benchmark** (TMI-I), exercise 1 was calculated with APROS 4.06 point kinetics model. In this calculation, the model was updated according the renewed specifications (figure 11). The calculated APROS results were compared with SMABRE calculations by VTT and with TRAC calculations by benchmark organizers (PSU) in report TOKE 6/1999 (Karppinen & Puska 1999). The main difference between APROS and SMABRE results is in the break flow rate. In SMABRE calculation, more liquid was carried with steam flow to the break and the steam generator dried out earlier. In APROS calculation, steam flow to the break lasted longer and about 7000 kg more steam was released through the break. The higher steam release in APROS calculation cooled down the plant longer. This resulted in lower core temperatures and in a power increase that lasted longer (figure 12).

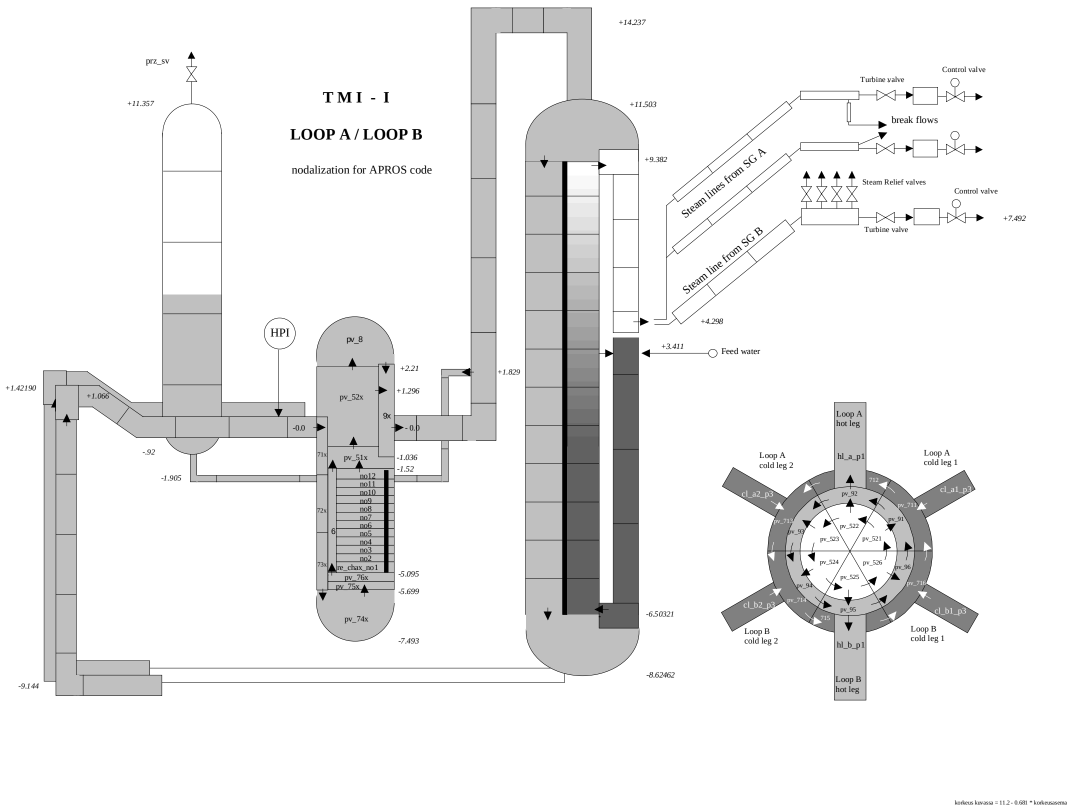

Figure 11. APROS model for TMI I.

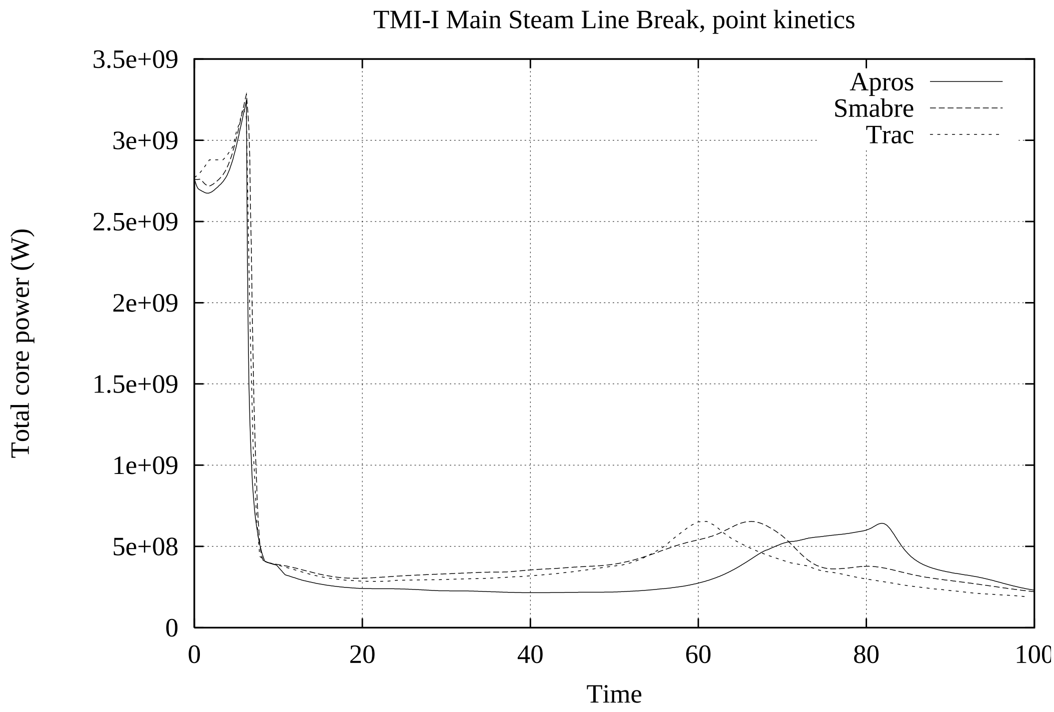

Figure 12. Power in the TMI MSLB point kinetics calculation.

A new APROS input for the simulation of PACTEL experiments has been created with the help of the GRADES user interface (Leppänen 2000). The model is more detailed than the previous APROS model. The model was built with 6-equation thermo hydraulic model and contains 454 nodes, 513 branches and 1478 heat structure nodes. The model validation against the PACTEL secondary side boil-off experiment LOF-10 and the small break LOCA experiment SBL-22 will be described in a separate paper of this seminar (Leppänen et al. 2000).

The **condensation model of APROS has been improved** by installing Chen interfacial heat transfer correlation and Wierow-Scrock model for non-condensable gas. The new condensation models were validated with PANDA Isolation Condenser and NOKO Emergency Condenser tests (Dreier et al. 1998, Karppinen 1999, Karppinen et al. 2000, Krepper et al. 2000). The improved model gave better results in the vertical test rig simulating the Passive Containment Cooler of the General Electric ESBWR plant consept. The Wierow-Schrock model is used to calculate a correction for condensation when there is non-condensable gases mixed with the vapour. The condensation efficiency was compared with the PANDA tests (figure 13).

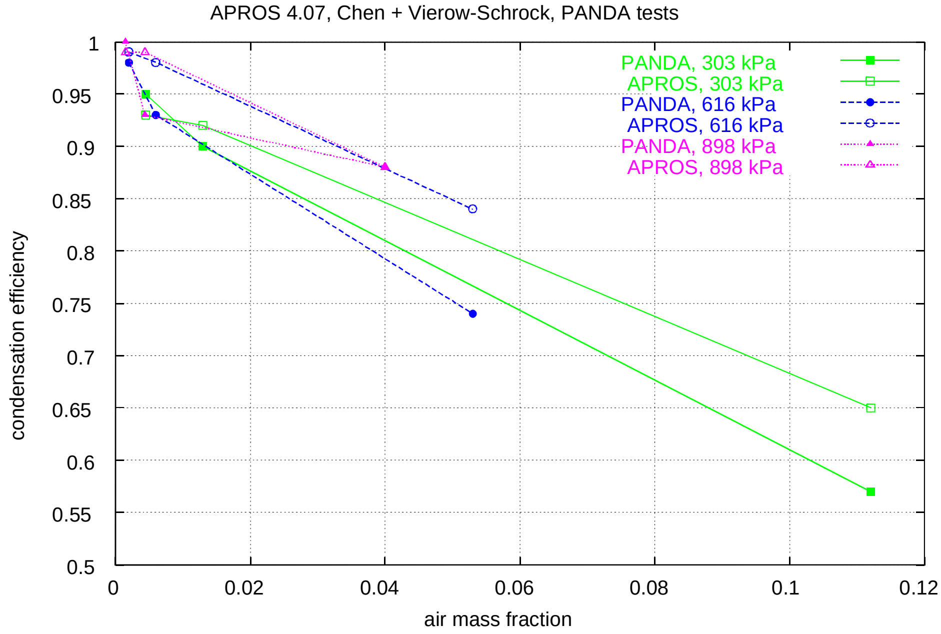

Figure 13. Results of the APROS simulations of the PANDA steam-air tests.

Thermal stratification of hot and cold water may induce leakages in pipes of power plants (Eerikäinen 1999). A test loop for carrying out **experiments to study thermal stratification** in a small diameter vertical tube has been designed and built (Puustinen 2000). The experimental set-up is based on the geometry of the connection line between hot and cold legs in the Loviisa NPP. In the test loop, a smaller (ID=50 mm) vertical pipe is connected to the bottom of a larger (ID=243 mm) horizontal pipe. Turbulent, hot (250 °C) flow in the larger pipe and cold stagnant fluid in the vertical pipe will cause cyclic temperature behaviour near the T-joint. In order to follow formation and movement of temperature gradient, the vertical pipe section close to the joint is equipped extensively. The loop is connected to the nearby PACTEL test facility, which produces heat and flow for the tests. In the first phase of the experiments, the location of the steepest temperature changes will be detected using different flow parameters in the pipes. In the second phase, a more detailed instrumentation will be used for determination of the temperature behaviour of the fluid and the structures.

The **collector header rupture experiments** PSL-10 and PSL-11 (Riikonen 1999) have been calculated with the RELAP5/MOD3.2.2Beta code. The main features of the RELAP5 model were based on the model by Riikonen (1996). A detailed model was added to simulate the primary-to secondary leak configuration in the experiments. The trigger logic was updated according to test procedures.

After the break opening, the calculated overall thermal hydraulic behaviour corresponded reasonable well to the experiments in both cases. The primary side started to depressurize, and the collapsed level in the pressurizer decreased as expected. The corresponding level rise was observed on the secondary side of the broken steam generator. Decrease of the primary pressure from the steady state level to the hot leg saturation pressure was predicted well in case of the experiment PSL-10 having the larger break size. In case of the smaller break experiment PSL-11, the calculated decrease in primary pressure and pressurizer level almost stalled when the circulation pumps started to coast-down, see figure 14. A probable explanation is that RELAP5 calculated too low circulation through the upper part of the reactor vessel. Hot water stored there reduced the calculated decrease of pressurizer level and primary pressure. The situation remained stable until the operators started cooling of the primary circuit by decreasing the secondary pressure in intact steam generators.

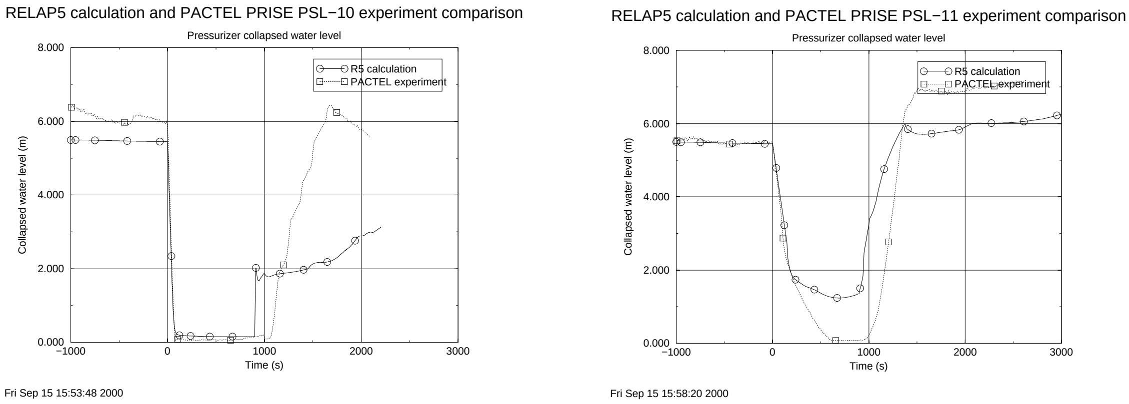

Figure 14. Comparison of the measured and calculated collapsed water level in the pressurizer in PSL-10 and PSL-11 experiments.

An extensive **reprogramming of the data acquisition system** and a full hardware/software **update of the process control system** of the PACTEL test facility is being done. The old UNIX based self-made measurement program was replaced with commercial Windows based software. The new data acquisition program helps in running tests by making the use of measurements more flexible. It is now easier to configure the data acquisition system according to the requirements of individual tests than before. The post processing of the measured data requires less work than with the old program. The new and more sophisticated process control system replaces the old programmable logic, which was getting obsolete. It was also becoming rather impossible to find spare parts for the old system.

Main objectives in 2001–2002

In the VVER related tests part of the project, the main objectives in 2001–2002 are as follows: A series of LBLOCA tests with the PACTEL facility will be carried out, where, for example, the influence of the break size on the maximum cladding temperature will be determined. A new condensation system has to be built and the scanning speed of the data acquisition system has to be increased in order to be able to do the tests. The results of the tests will be used for code validation. A second test series focusing on the behaviour of horizontal steam generators in the presence of non-condensable gases will be planned and carried out. All three primary circuits of the PACTEL test loop will be used. Helium acts as a non-condensable gas. The results will be used for the assessment of APROS non-condensable gas model.

In the code validation subproject, the main objective is to continue APROS validation by taking part in the international benchmarks, by using possible plant transient data and by utilising appropriate PACTEL test results. The new PACTEL input model created earlier in the TOKE project will be used in the calculations. RELAP5 calculations will be continued as part of the CAMP agreement.

The separate effect tests are aimed at producing more thermal stratification measurement data with different configurations, for example, in horizontal, small diameter tubes. The tests will be carried out in co-operation with the STIN project. In addition, experiments addressing the problem of water hammer phenomenon will be started. The goal is to collect data with high-speed transducers and data acquisition for the assessment of structural analysis computational methods. The produced data is utilised also in the STIN project.

Applications

PACTEL integral test results can be used to develop accident prevention strategies in VVER nuclear power plants. New calculation models developed in accordance with test results and integrated into APROS code will improve the applicability of the code for accident analysis.

The results of the thermal stratification separate effect tests will be used as boundary conditions for thermal fatigue and structural integrity analyses. Thermal stratification tests create new and valuable co-operation between different branches of nuclear safety research i.e. between thermal hydraulic and structural integrity studies.

PACTEL test results are essential as a Finnish contribution to international programmes, such as CAMP, and can also be used in EU projects which aim at enhancing safety of the Eastern European reactors. Beyond the FINNUS programme, the PACTEL facility is used in the ALWR experiments and for the needs of the Finnish utilities.

Note: A special report on the TOKE project in chapter 5.8.

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3.4 Modelling and simulant experiments of severe accident phenomena (MOSES)

The safety of the Finnish nuclear power plants against severe accidents has been enhanced by several plant modifications and extensive research efforts during the past decade. However, significant uncertainties still remain in some areas that are important to the Finnish nuclear power plants, such as coolability of core debris, failure mode of the pressure vessel and fission product behaviour, particularly in the reactor coolant system. In order to address the identified research needs, the project focuses on reducing uncertainties and getting better insights of pressure vessel lower head response to core melt attack, coolability of debris beds, chemistry of fission product iodine and containment thermal hydraulic loads.

Main objectives in 1999–2000

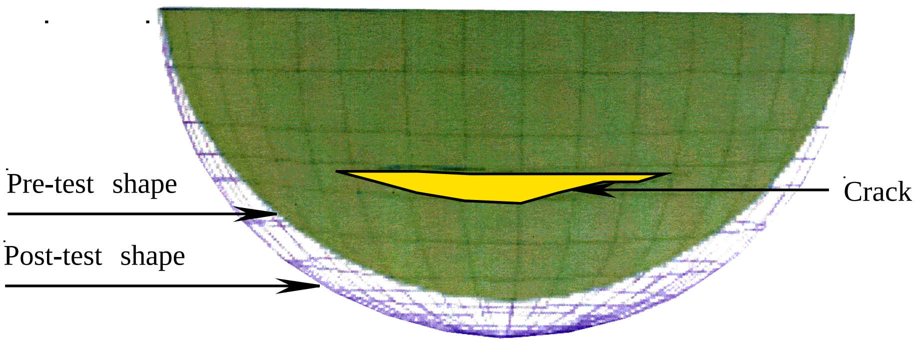
In the area of in-vessel phenomena the main goal of the MOSES-project has been to complete development of a pressure vessel lower head creep model and validate it against $1/5^{th}$ scale lower head creep tests performed at Sandia National Laboratory. Another major task has been the start-up of own experimental work on severe accident thermal hydraulics. The first phase of the test series was to examine heat transfer in dry, oxidic particle bed at high temperatures. These tests were tailored to produce data for validation of the granular bed heat transfer model in the PASULA code. The second phase of the VTT tests are studying coolability of granular debris by measuring dryout heat flux in simulant material particle bed. The pre-studies and the construction of the test facility are to be completed during 2000. In the area of fission product behaviour the main efforts are to follow-up large-scale Phebus-experiment programme and to investigate the behaviour of organic iodine in the containment. The Phebus-tests, performed in France, study in-vessel melt progression and fission product behaviour. The task with behaviour of organic iodine is to find and evaluate methods to prevent organic iodide formation in the containment during a severe accident. The work in MOSES-project comprises a literature study on organic iodide formation and search for suitable reactants to retain it in containment water pools. This work is also part of the NKS/SOS-2.3 project. The last major target area is the assessment of hydrogen detonations in a BWR reactor building. The task comprised first a literature study on detonation phenomenology and a rough estimation of pressure and impulse loads acting on reactor building walls during an explosion. On the basis of the first study, more accurate 3D-analyses with DET3D code are being performed during the year 2000. The work on hydrogen detonations is also part of Nordic SOS-2.3 project.

Main results in 1999–2000

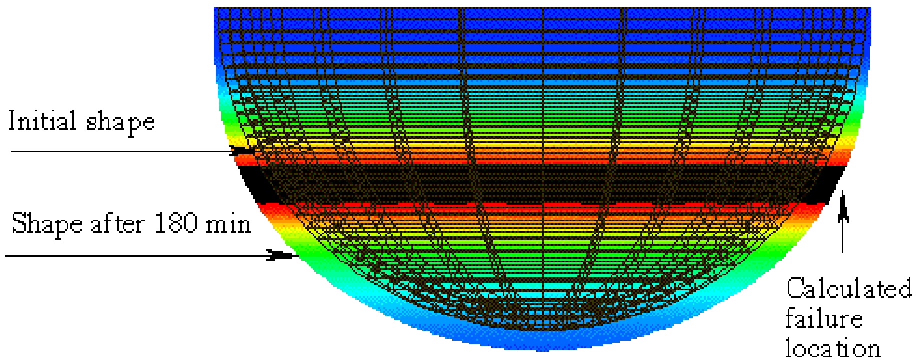
The work on **pressure vessel failure mode** focused on calculation of Sandia LHF-3 creep rupture test with PASULA/FEM model (Ikonen 1999, Lower Head... 1999). The test vessel was a 1/5th scale hemispherical mock-up of a pressure vessel lower head cast of American standard pressure vessel steel. The lower head was pressurised up to 100 bar and simultaneously heated uniformly. The PASULA prediction of the wall strains were slightly smaller than the measured ones. However, the shape of the strain profile was well predicted, with the maximum at the angle of 30^{o}, where also the largest displacements were measured. The vessel failure time was estimated by two different criteria: the rule of 20 % cumulative effective strain and the linear life fraction rule of Larsson-Miller type approach. The linear life fraction rule predicted much longer times for failure than actually occurred in the test. The criteria of 20 % cumulative strain, however, predicted the failure time well: 165 minutes to be compared with the 177 minutes measured in the test.

The VTT dry bed test rig was designed and constructed for **heat transfer experiments**. The test rig involves an instrumented and heavily insulated test furnace containing a particle bed made of alumina balls (Holmström & Auerkari 2000). The vertically aligned particle bed had a diameter of 100 mm and was 200 mm high. The heating of the particles was realised with a spiral resistance heater at the top of the bed. The particle bed was placed in a thin ceramic tube surrounded by 100 mm thick ceramic wool-type insulator. Temperatures were measured at 9 locations in the particle bed and at 12 locations in the insulator. The test facility is shown in figures 15–17. The spherical, uniform alumina particles used in the bed were about 6.7 mm in diameter.

Five successful testing runs were completed. The power was increased stepwise in each test after reaching a steady state in heat transfer with each power level. The first two tests were run with air in the particle bed, but as the temperatures increased it was necessary to add small (2 l/min) Argon gas injection to the particle bed for better survivability of the thermocouples. At the maximum heating power applied (418 W), the maximum achieved temperature before failure of the heating element was 1270 ^{o}C at a distance of about 10 mm from the surface of the heating element.



Schematic illustration of the LHF-3 lower head after the test [2].



PASULA prediction of the LHF-3 lower head at the end of the test.

Figure 15. Measured and calculated lower head distortions during a creep test LHF-3.

Figure 16. VTT test rig for dry particle bed heat transfer measurements.

Figure 17. Al_2O_3 particles used in dry bed experiments.

As an example figure 18 illustrates the measured temperature histories of thermocouples located on the centre line of the particle bed.

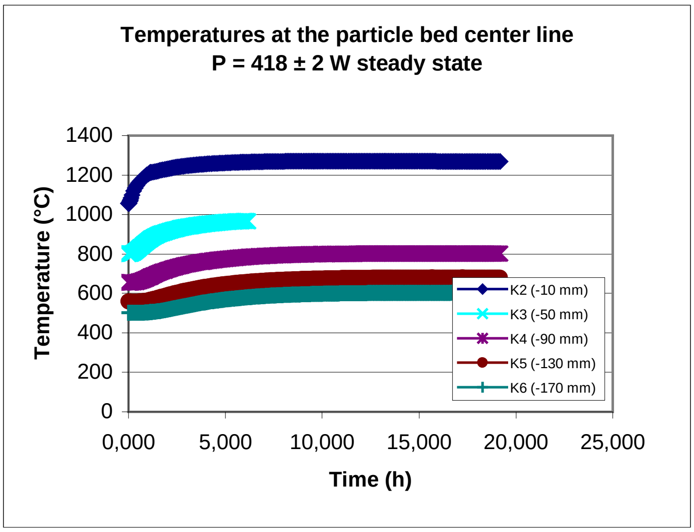


Figure 18. Measured temperatures in the particle bed with maximum applied heating power. Thermocouple K3 failed at about 6 hours.

The dry bed heat transfer experiments were calculated with the PASULA and a specially developed BEDEXP codes (Ikonen 2000, Miettinen 2000). The calculation efforts were complex due to the fact that the material property data provided by the manufacturer were not accurate enough for scientific applications. Furthermore, the aluminium oxide particles had some internal porosity that changes the material properties in comparison to the solid Al_2O_3 properties. BEDEXP analyses supported the assessment of correct material properties in PASULA calculations. The analyses of VTT dry particle bed experiments suggest, that the numerical model developed for particle effective heat transfer coefficient in the PASULA code is capable of predicting heat transfer phenomena in a dry particle bed with good accuracy, once the solid particle material and cover gas properties are known. Figure 19 shows the calculated and the measured thermocouple temperatures in the particle bed and insulation.

Moreover, the lesson learned from this analytical exercise was that even simple radiation/conduction heat transfer phenomena may be complex to measure, and that the numerical modelling may easily suffer from inaccurate material properties. However, the performed study also shows, that with sufficient number of temperature measurements from carefully designed locations one can obtain a lot of information from the system components that can further facilitate model assessments.

The design of the wet particle bed **dryout heat flux experiments** have been started. A literature study was performed on core melt fragmentation to aid in defining of a representative particle size distribution for the simulant MOSES2-experiments (Lindholm 2000). The available experimental data on fragmentation suggests that the formed particle bed would have a high porosity 40–60 % and the particle bed would have an average size of 3.5 mm. With a simple assessment the amount of core debris that may undergo energetic interaction with coolant forming finer particles, it was concluded that 10 cm additional finer particle layer may settle on top of the coarser bulk debris. A literature study was also performed on previously conducted dryout heat flux measurements in particle beds (Hyytinen & Lindholm 2000). The existing data base on dryout heat fluxes covers a wide range of particle sizes and both homogeneous and stratified bed configurations. On the basis of first, simple dryout heat flux calculations, a bed with average particle size of 3.5 mm would be coolable by top flooding.

A novel problem of hydrogen accumulation into a high concentration cloud in the Olkiluoto reactor building was brought up. In such a case the threat to containment integrity would come from external explosions that might damage the containment penetrations. For addressing this issue, a literature study was performed on **detonations of hydrogen-air-steam mixtures** (Silde & Lindholm 2000). Firstly, estimates on detonation loads were obtained by applying theory of strong explosions with instantaneous, point-wise energy release and shock wave reflections from a structure.

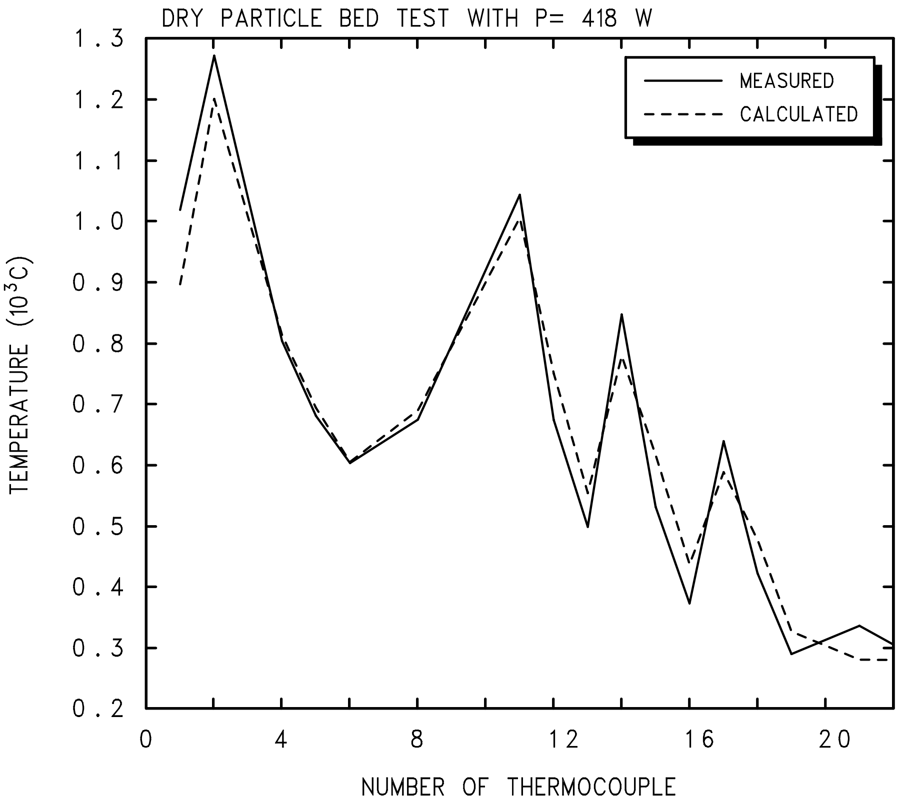


Figure 19. PASULA predictions and measured temperatures of thermocouples in particle bed and in insulation in the test with maximum heating power.

This theory gives conservative, rough estimates of the pressures and impulses of the first shock wave reflection from the wall. The assumed hydrogen distribution in the studied reactor building room is based on earlier studies (Manninen et al. 2000). The detonable hydrogen mass was assumed to be 1.5–3 kg, and this caused a pressure spike of 13–39 MPa, with corresponding impulses being 2–9 kPa-s. These estimates, however, do not take into account the gradual energy release during propagation of the combustion. Neither do they account for the multiple, 3-D reflections and focusing of shock waves in corners. A more detailed analysis taking the detonation dynamics accurately into account have been performed with the 3-D code DET3D developed at Forschungszentrum Karlsruhe (Silde & Redlinger 2000). This work has been performed as part of the Nordic nuclear research programme NKS/SOS-2.

Another topic, that has been worked upon in Nordic collaboration, is the **behaviour of organic iodine**. The Nordic project, ordinated by VTT Chemical Technology, comprises both experimental and analytical studies. The contribution of the MOSES project to these studies was to gather available information of the methods to prevent a source term of methyl iodide during a severe accident (Karhu 1999). The most widely studied methods for nuclear power plant applications include the impregnant carbon filters and alkaline additives and sprays. The formation of elemental iodine, that could react further producing organo iodides, is minimal in alkaline solutions. Hence, strong basic caustic chemicals, such as NaOH, KOH and LiOH, are readily available in most nuclear power plants. To avoid the corrosiveness of these chemicals, different buffer solutions, like borate and phosphate buffers, are considered as candidates for pH-control in nuclear power plants. Filters are commonly used to remove gaseous iodine. Impregnants, such as TEDA and KI, are used to improve the performance of the filters. TEDA has shown high affinity toward methyl iodide, but it is not stable in acidic conditions and it has a low ignition temperature. Therefore, there is a need to develop more reactive and stable impregnants. Most promising candidates are recognised to be different transition metals, like silver and zinc. The next phase of the work is dealing with a more through investigation of reactions of candidate materials, which will be ultimately tested in laboratory scale.

Main objectives in 2001–2002

The key goals of the second half of the MOSES-project are the performance of the particle bed dryout heat flux experiments. The first experiment will be performed with a particle bed with a representative particle size distribution, when steam explosions are not taken into account. The second experiment is planned to have an additional 10 cm layer on top of the base bed with finer particles. This will address the effect of an order of magnitude smaller particles, formed in an energetic interaction, on dryout heat flux. A calculation tool will be developed and applied for numerical post-test analyses.

A computer code interface will be built between hydrogen detonation code DET3D and ABAQUS structural analysis code. The interface will facilitate an automatic transport of calculated 3-D pressure loads to the input of the ABAQUS code. Furthermore, detonation pressure loads on system 321-pipeline penetration in Olkiluoto reactor building will be calculated with DET3D code. The integrity of the pipe penetration under these circumstances will be analysed with the ABAQUS code in the STIN project. This work will be part of the Nordic collaboration.

An new research task that will start in the frame of Nordic collaboration will be the analysis and possibility of recriticality of a BWR core, when molten control rod material is slumping into the water pool in the pressure vessel lower head. The goal is to evaluate, if the melt can fragment violently causing rapid vaporisation that will push a liquid slug of unborated water to the core region without absorber material. The work-share of MOSES-project will be the assessment of recriticality.

In the area of pressure vessel integrity, the follow-up of the international OLHF-project will be continued and new lower head creep rupture tests will be analysed with the PASULA/FEM-model.

Moreover, a number of significant international research programmes in severe accident area, like MASCA, CSARP and MACE continuation, will be participated.

Applications

The development and validation of PASULA/FEM-model provides an applicable tool for assessing the behaviour of the Olkiluoto reactor pressure vessel lower head in a case, where the core melt is unable to discharge through the lower head penetrations. Such a situation may occur if the relatively thin penetration tubes are blocked by refrozen metallic melt. A similar creep rupture analysis has been performed for the Loviisa plant earlier in the REVISA EU project.

The dryout heat flux experiments are readily tailored for the likely situation in the Olkiluoto reactor pedestal during a severe accident. With supportive calculations an assessment can be performed, if a rubble bed in the pedestal is coolable with the current accident management measures. However, international research and experiments are needed to evaluate reliably and efficiency of the concept to cool a melt pool from the top.

Note: A special report on the MOSES project in chapter 5.9.

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4. Risks: Main goals and results

4.1 Fire safety research (FISRE)

Fire safety studies include influence of smoke and heat from fires on electronics and programmed logic because of modernisation of the existing plants. Fire risk analysis has not yet developed mature enough for living PSA; therefore, fast risk analysis methods starting from the scenarios contributing strongest to existing plants are being developed. First very simple fire models are used for sensitivity types of studies with an emphasis on plausible probability distributions of input variables. Guided by the obtained results more sophisticated methods will be developed later for sensitive parts of risk analysis link. Likewise the reliability of active fire protections measures are an object for further studies.

Main objectives in 1999–2000

For the first period of FISRE-project the main theme is to improve the calculation tools used to support PSA-analyses. This goal is approached on three fronts: (a) experiments and modelling on hardware, (b) software development and assessment, as well as (c) processing of statistical information on the availability of active fire protection equipment.

(a) Test series on heat effects on instruments was concluded. The acute influence of smoke and heat from fires on electronics and programmed logic is an interesting research area worldwide, because of modernization of existing plants. The goal was to investigate what types of disturbances smoke could cause through accumulation of soot on printed card boards.

(b) Fire spread from a cable tray to the neighbouring tray is still a major risk scenario in a NPP, where various redundant trains are not separated physically by fire rated partitions. The goal was to develope a simple fire risk analysis method for sensitivity analysis of a fire in a cable tunnel. A generic calculation platform should be built for estimation of the probability distribution of the target event. Since the probability distributions of input parameters are mostly unknown, stepwise sensitivity estimations should be made to guide efforts of input data collection. Fire risk analysis will be developed towards living PSA. Practical calculation tools with input data bases, as well as realistic input data distributions will be developed step by step.

(c) Statistical studies was to be made on reliability of sprinkler and fire alarming systems both in nuclear and non-nuclear installations in Finland. The task was initiated by international literature study, which revealed that only obsolete, questionable generic data was available. The quality of available Finnish statistical information was explored, which showed that scientific treatment of data might be worthwhile for sprinklers and fire alarm systems. Then studies were undertaken to extract quantitative reliability data on subsystems and components. Since the population on Finnish NPPs is very small, non-nuclear data was added to shead light on the possible malfunctions. Data from Swedish NPPs are to be added later.

Main results in 1999–2000

Effect of smoke and heat on electronics

The goal of the first task was to study performance, damages and failures of the **equipment of NPPs at elevated temperatures** beyond the normal operation limits. Heat transfer into the measuring and electronics units of a working pressure transmitter as well as a valve actuator with electric motor was studied using theoretical and experimental methods. Data reduction of earlier experiments was completed and the report finished (Björkman & Keski-Rahkonen 2000).

For equipment contained within metal boxes a very simple general estimate of the critical heating time was obtained. In case of fire a sudden change of ambient temperature ΔT_a occurs. The critical temperature ΔT_{cr} above normal ambient temperature is the maximum temperature, where the instrument still functions. The time t_{cr} when this temperature is reached is called the critical time span. This time is obtained from

$$\frac{t_{cr}}{\tau} = -\ln\left(1 - \frac{\Delta T_{cr}}{\Delta T_a}\right) \tag{1}$$

where τ is the heating time constant of the instrument box. The value of τ can be determined experimentally using similar experiments that we carried out here. However, the universality of this result becomes valuable, because a rough value of τ, accurate enough for risk analysis purposes, can be calculated from the mass and physical dimensions of the instrument.

A few smoke exposure experiments on mockup electronic circuits were made. A quantitative model of **smoke deposition on electronics** was proposed. Different measurements on the operation of electronic circuitry were made to explore possible faulty operation. Fast change of insulation resistance was observed, but other measured effects were small. Finally, a model was written to describe loss of insulation, as well as for requirements of a protective layer. A real commercial circuitry with a protective laquer layer on circuit board worked without fault during smoke exposure, (Mangs et al. 2000). (See also chapter 5.10.)

Modelling of fire scenarios for PSA

The main goal is a simple **fire risk analysis** method for sensitivity analysis of a fire in a cable tunnel. It will be made concentrating on the influence of the unknown probability distributions of input parameters. Fire risk analysis will be further developed towards living PSA. Practical calculation tools with input data bases, as well as realistic input data distributions will be developed.

The model is a user platform interfacing with various fire models of different complexity and a commercial risk analysis package @RISK. For demonstration, a sensitivity analysis was made for a small group of unknown variables to show that the tool can be used to distinguish the important variables from the less important. The distributions of the target cable loosing times were produced with three different mean values of the RHR (rate of heat release) growth rate, (Hostikka & Keski-Rahkonen 2000).

The current method can be applied with two-zone fire models. It offers a flexible framework for widening to meet the requirements of real, industrial scale applications. The most important aspects of the work going on during the current year are:

- A comprehensive sensitivity study involving all the relevant input variables.

- A quantification of the true statistical distributions for the relevant variables.

In the scenario a cable tray is ignited, and probability of ignition of a neighbouring reduntant cable train is calculated. Figure 20 depicts behaviour of these probability curves. Fire growth rate of the first ignited item is assumed parabolic in time. Probability values are plotted for three different values: 1, 2 and 3 W/s^2. Other distributions of the input parameters were not yet necessarily realistic at this phase but were selected through engineering judgement. Figure 21 shows the sensitivity of different input parameters in the ignition measured as rank correlation coefficient. The given three parameters, rate of heat release, cable height and source height are likely the three mostly contributing factors even for realistic input distributions. Therefore, efforts to get real data are concentrated first on these variables.

Two conference presentations of the task has been prepared (Hostikka et al. 2000, Keski-Rahkonen 2000c).

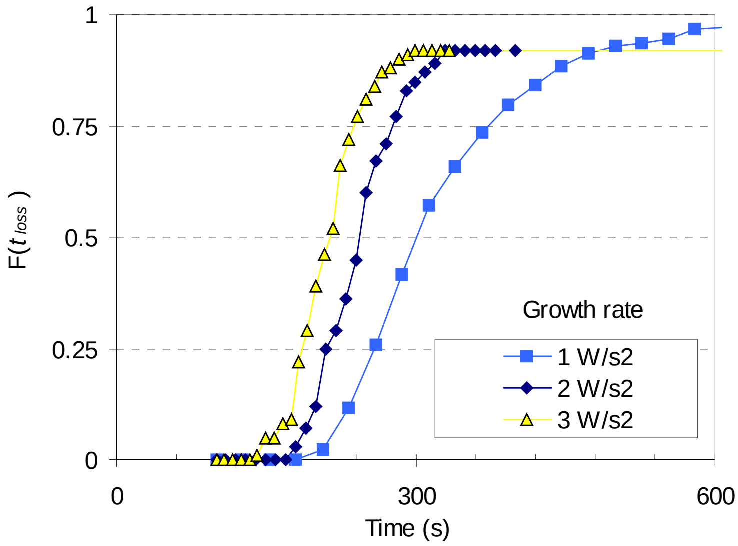

Figure 20. Sensitivity of the cable loss time distributions to the RHR growth rate.

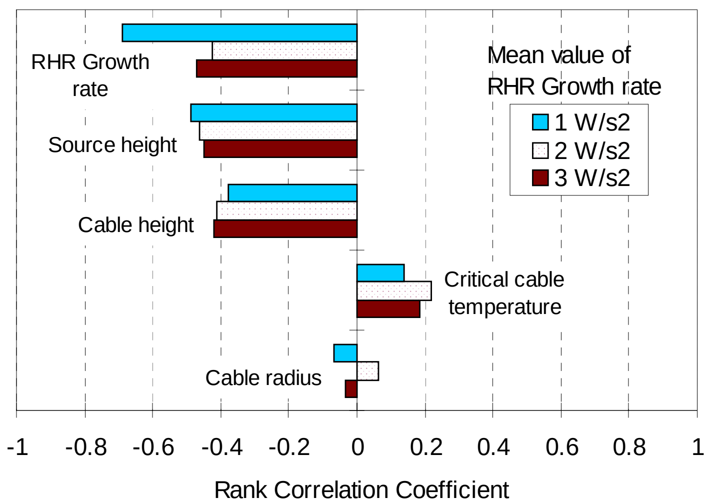

Figure 21. Sensitivity of the cable loss time to the different input variables.

For early warning of fires **smoke detectors** are very sensitive devices. In a growing fire the density of smoke inside the detector is not the same as outside in the free ceiling jet. Therefore, the detector does not alarm at the time an ideal detector in a free air would alarm. This is known as time lag of the smoke detector. Heskestad (1975) drafted a theory using dimensional analytic arguments for the time lag Δt of a products-of-combustion fire detector

$$\Delta t = \gamma\, l \,/\, \bar{v} \qquad (2)$$

where $\bar{v}$ is the mean convective flow velocity around the point detector, l the characteristic length scale of the detector, and γ a nondimensional coefficient characteristic for the detailed geometry of the detector. According to Heskestad (1975) Equation (2) is valid presuming 'viscosity effects are not considered important'.

Brozovsky (1991) measured time lag for a detector shown as dots in figure 22. Plotting Heskestad's correlations according to Equation (2) would yield curves shown in dotted lines in figure 22. For the lower curve $\gamma \boldsymbol{l}$ = 0.8 m, and for the upper curve 8 m. It is clear from the figure that these predictions are in contrast with experiments. No value of $\gamma \boldsymbol{l}$ would fit them with experimental data. On the contrast, simple data manipulation showed, an equation of the form (Keski-Rahkonen 2000a)

$$\Delta t = \gamma \boldsymbol{l} \,/ (\bar{v} - \bar{v}_0) \qquad (3)$$

would yield a plausible fit on Brozovsky's data (figure 22). Here the numerical values of the parameters were: $\gamma \boldsymbol{l}$ = 0.8 m and $\bar{v}_0$ = 0.075 m/s. Without any deeper physical insight into the problem, comparing Equation (3) with Equation (2), it is easy to reach a conclusion: At low velocities viscosity effects are not unimportant.

A fluid mechanical study is going on to calculate lag time from the basic fluid dynamic principles and the dimensions of the detector (Keski-Rahkonen 2000b). Furthermore, from published experimental data statistical distributions of the response time of the detectors were derived for input data bank of the calculation tool.

U.S. Nuclear Regulatory Commission started 1999 an **International Collaborative Project to Evaluate Fire Models for Nuclear Power Plant Applications (ICPEFMNPPA)**. The goal is to assess fire development (zone)models. It was decided to participate in the project at low volume. A project preparation meeting was held at IPSN, Fontenay-aux-Roses, France, June 19–20, 2000. Comments for a benchmark problem were made, and two presentations of our work made (Keski-Rahkonen 2000c, d).

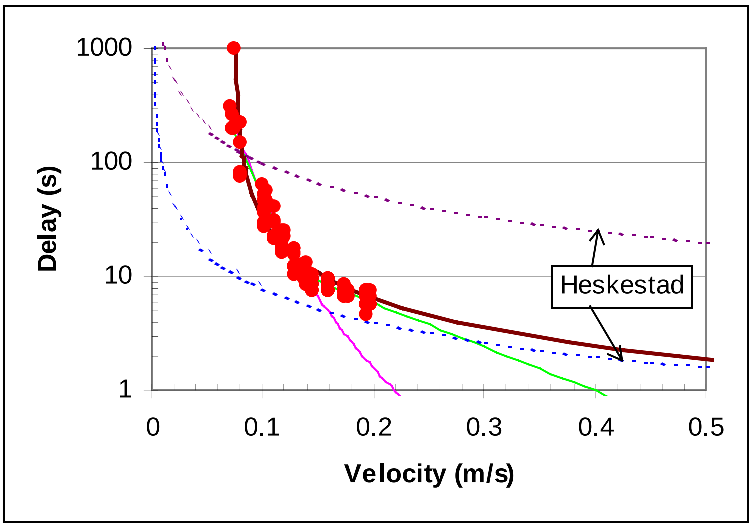

Figure 22. Entry lag time dependence on flow velocity past a detector: dots (experimental data), and thin full lines (exponential fits), (Brozovsky 1991). The delay time according to Equation (2) is plotted by dashed lines (Heskestad) for two values of $\gamma \boldsymbol{l}$ *= 0.8 m (lower curve) and 8 m (upper curve). The heavy solid line is the lower line delayed by* $\bar{\mathbf{v}}_0$ *= 0.075 m/s according to Equation (3).*

Active fire protection equipment

The goal is to provide critically reviewed numerical values for reliability of selected active components like fire detectors and sprinklers using statistical data from nuclear and possible non-nuclear installations. A literature study showed that very little on reliability of these systems is available internationally. One of the useful findings was data for distributions for responding sprinkler heads (Rönty et al. 2000). A fair fit (figure 23) of the observed probability ***f(n)*** of ***n*** heads operating was obtained using a relationship

$$f(n) = n^{-s} / \zeta(s); \quad s > 1; \quad n = 1,2,3 \ldots \tag{4}$$

where **s** is an exponent determined from statistical data, and the normalizing factor $\zeta(s)$ is the Riemann zeta-function.

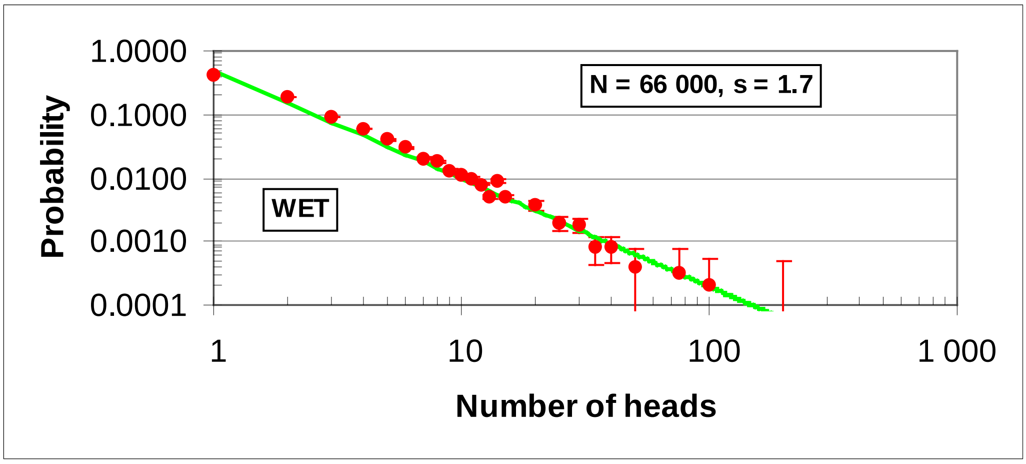

Figure 23. Probability distribution for the number of sprinkler heads responding to fires.

Reliability of fire detection devices was started by inspecting data from non-nuclear detector inspections (SPEK), because this material provides a large population to show occurence of even rare events. It was extended to data from our NPPs, which will be evaluated to extract numerical values of reliabilities of the system/components. The work is still going on.

Reliability of sprinkler extinguishing systems was also started using non-nuclear data (SVK) to get larger populations. Observed fault types were counted, and simple fault tree modelling of the systems carried out. Statistical data from our NPPs were also collected.

A key question for calculating reliability data is to know the size of the component population as a function of time during the years inspected. Since it was not known prior to our study special laborious efforts were needed to estimate it. From nuclear installations real drawings and component lists were obtained. These were used to count the number of different components, and length of pipelines used. Additionally, floor areas of rooms were recorded to estimate component densities. This is needed to bind sprinkler installation reliability to fire frequencies, which are known per floor area (Rahikainen & Keski-Rahkonen 1998).

Drawings were obtained from in 102 non-nuclear buildings, and floor areas as well as number of components of sprinkler installations were counted. Cumulative distribution of protected floor area is plotted in figure 24. Curve fitting using a cumulative Weibull distribution

$$F(x-x_0)=1-exp\left\{-\left[(x-x_0)/\beta\right]^{\alpha}\right\} \quad \alpha>0,\beta>0, \quad 0\leq x_0\leq x\leq\infty \qquad (5)$$

where **x** is the floor area [m^2], yields a good fit with parameters: $\mathbf{x_0}$ = 194 m^2, $\boldsymbol{\alpha}$ = 0.8, and $\boldsymbol{\beta}$ = 8 000 m^2. This distribution for protected floor area, as well as similar distributions for other relevant parameters allow estimating the population sizes of different sprinkler installations and number of components from known number of sprinklered buildings. The work of extracting of these data is going on.

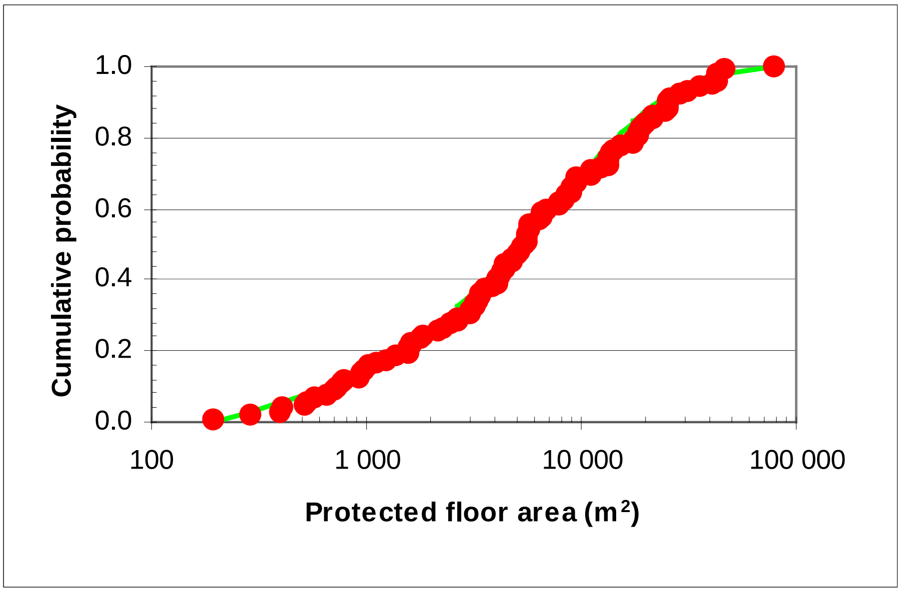

Figure 24. Cumulative distribution of the floor area of sprinklered buildings (dots), and a three parameter Weibull distribution fit (line) on the data.

Conclusions

Effect of sudden high temperature on control equipment has been assessed. A quantitative model for calculating critical heating time is proposed and validated for two specific device types. Acute effect of smoke on control electronics has been explored. Model for soot accumulation and deterioration of insulation resistance allows a quantitative tool to estimate requirements for protection of electronic circuitry against soot by using various coatings. The model shows as expected, that modern electronics, like that used for today's programmable automation circuits, is in principle more vulnerable for smoke and soot than the conventional analog control circuitry used at the time our nuclear power plants were built. As for insulation resistance our experiments showed that use of protecting coatings could reduce the problem to a tolerable limit.

A calculation platform has been written, which allows estimation of probability distributions for ignition of the next target in a cable tunnel. Input data bank for real distributions has been started. Although applied to a specific scenario the platform is general allowing use of different targets, and calculation tools. This calculation method is needed for quantifying fire simulation prediction for PSA probability estimates.

Extensive data mining from active fire prevention devices was started to extract reliability values first to sprinkler and fire detection systems. Initial evaluation of the available data showed promising. Quantitative modelling, and calculation of system and component reliabilities is going on.

Note: A special report on the FISRE project in chapter 5.10.

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4.2 Programmable automation system safety integrity assessment (PASSI)

The licensing of software-based systems is a difficult task. During the licensing process, different kind of evidence on the system under consideration is collected and analysed in order to make sure that the system fulfils the safety requirements set by the authority. The evidence consist of information on the systems development environment and tools, the systems design documents, data from the development process (e.g. quality control data), different metrics measuring the properties of the software, data from system testing, and operating experience data.

The Finnish nuclear safety authority STUK has set a quantitative reliability target for safety functions, and thus one problem in licensing of software-based system is the quantitative demonstration of the systems reliability (YVL 1997). Due to these requirements, the view adopted in the PASSI-project is that of reliability analysis. The project itself is a continuation of earlier OHA-project (“Programmable automation systems in nuclear power plants"). OHA-project studied e.g. the diversity and testing principles and the statistical reliability analysis of software-based a systems. Furthermore, the problems connected with software-based systems in NPP PSAs, and the features of the whole licensing process were considered in the OHA-project (see. Haapanen et al., 1997, Korhonen et al. 1997, Pulkkinen, 1997, Korhonen et al. 1998).

Main objectives

The PASSI-project concentrates on quantitative reliability analysis of the software-based systems. One aim is to develop, demonstrate and validate reliability assessment methods. Our starting point has been the fact that quantitative probabilistic reliability estimates are predictions, which are based on the totality of the evidence about the safety of the system. Thus, the problem is to collect the evidence, model the impact of different pieces of evidence on system reliability, combine and weight the pieces of evidence in a consistent way, and finally produce a reliability estimate. Further, in characterising the evidence, also the related uncertainties must be identified and analysed.

Three tasks are connected to the reliability analysis: Reliability assessment methods (REL), Case study on operating experience analysis (OPEX), Benchmark exercise on analysis of software based systems (BE-SECBS). In addition to these tasks, a specific task on ageing of I&C components (ICAGE) is included in the project following the recommendations of the strategy group of the FINNUS program.

Main results in 1999–2000

Reliability assessment methods (REL)

The work in this area concentrates on the development and assessment of methods and tools needed for the handling of different kinds of quantitative and qualitative evidence on the high quality of the system and its design process, when the reliability and safety of that system is assessed. The emphasis is on the quantitative reliability modelling, but the connections to other kinds of modelling are considered, too.

The Bayes networks have been selected as the basic tool for reliability analysis. This is due to the possibility to model in a transparent probabilistic way the connections and dependencies of different pieces of evidence. The Bayes network structure for analysing the operating experience data was established in the project (see Helminen, 2000, and Pulkkinen & Helminen, 2000). The model can be applied in analysing the data from both a specific software-based automation application and its earlier versions as well as from the operation of the basic system (i.e. commercial automation system platform). The Bayes networks have been used for this purpose also by other researchers (see Littlewood & Fenton, 1996, and Dahll & Gran, 2000).

In addition to the structural specification, the work has included the specification of the parameters of the Bayes network model and their probability distributions. Along with the hard data from tests or operating experience, this requires extensive use of expert judgement. Principles for implementing expert judgement into specification of parameters and their distributions are under investigation. The work leans largely to the research made at VTT Automation in other connections (see Pulkkinen, 1994, and Pulkkinen & Holmberg, 1997).

The ”full Bayes network” model requires the implementation of "soft" evidence from the systems development and other types of safety arguments into the analysis. In the model developed in the project, this was made implicitly by taking it into account in the prior distributions of the model. However, in order to make the model practical and transparent, the work to describe this kind of evidence has been started.

In connection with the model development, some numerical experiments with Bayes networks were made using a share-ware computer code BUGS. It appeared that although BUGS algorithms performed satisfactorily, it may be advantageous to develop new algorithms, which utilise the specific structure and distributional assumptions made in the model used in the project (see Helminen, 2000).

The results from the Bayes network model are rather abstract, and they should be interpreted carefully. The principles to interpret and use the results in licensing of software based systems will be concluded later in the project. Then also recommendations for the use of the model will be given. This is connected strongly with the practical case studies, which will be made in the other tasks of the project.

The Bayes network modelling approach looks at the software based systems partially as a black box, which may treat certain types of evidence in too simplistic or conservative way. Thus, other approaches taking into account the requirement from the process to be controlled by the software-based system, are needed. This was emphasised also by the strategy group of the FINNUS-programme. To fulfil this need, the project considers possibilities to use the expertise from simulator development and other process analyses.

Case studies on operating experience and methodology benchmark (OPEX, BE-SECBS)

In order to test the practical applicability of the Bayes network model, it is planned to be applied in suitable case studies. Preliminary discussions on the programmable components for the case study have already been started (software controlled relays, ABB). The tasks of this subproject are

- selection of case for pilot study
- definition of the data (operating experience, design process data, etc.) to be collected
- collection of the data
- classification and qualitative analysis of data
- specification of the Bayes network model for the case study
- estimation of failure probabilities using the model
- reporting and conclusions.

The work on this subtask has been started.

In 1999, the preparation of project plans for an international benchmark study was defined as a task of the PASSI-project. The work resulted in a proposal for EU-shared cost action (BE-SECBS), which was submitted and accepted. Participants in the BE-SECBS-project are STUK, VTT, ISTec (Germany), IPSN (France), JRC-ISIS and Siemens (Germany). The project will be started in 2000. The objectives of the benchmark are:

Ageing of I&C equipment

Originally, the task of the ageing analysis of I&C equipment was planned as a part of FINNUS/AGE-project. However, it was identified later that a pre-study on I&C ageing could better be performed as a part of the PASSI-project in the year 2000. The main tasks of the pre-study are

- review of the earlier I&C ageing analyses (both nationally and internationally)
- identification of the need for ageing analyses
- identification of various types of ageing to be considered later (important groups of components, ageing trends, obsolescence, maintenance related problems etc.) and the approaches for their analyses
- identification of needs of power utilities and authority
- identification of national expertise on different ageing phenomena
- plan for an I&C ageing analysis project.

The ageing problems, research needs and the national expertise for solving I&C ageing problems were identified by a literature survey and interviews with power utilities, STUK, and research institutes. The interviews are now being reported.

According to the surveys both the safety authority and power utilities have made studies and developed ageing management approaches also for I&C equipment. Many kinds of I&C equipment and their ageing related failure modes have been covered by these studies (e.g. cables, relays, components of reactor protection systems). However, the main problem seems not to be the ageing of the components, but obsolescence of the technology. Majority of the I&C systems are based on old technology, the availability of which gets more and more difficult. Along with this, the expertise on old technology diminishes, and the utilities have to replace the old components and systems with new, in many cases programmable technology.

An ageing related problem with new technology is the rapid development of electronics. New production methods replace the old ones continuously, and the new component versions or technologies are introduced yearly. The life-cycle of the components shortens, and the producers of the electronics may have difficulties in guaranteeing the availability of components. This must be taken into account when system renovations are planned.

Another issue encountered in the interviews was the need for systemic or more wholistic view on ageing management and system renovations. This would help in identifying most important and safety critical ageing phenomena, as well as in organising regulatory work on ageing.

Conclusions

The PASSI-project has concentrated on quantitative reliability analysis of software-based automation systems. The approach adopted is based on Bayes network, which provide a modelling tool for transparent description and combination of the evidence behind the reliability estimation. The approach is a kind of black-box model, which does not explicitly take into account the information on the structure of the software. However, these features are taken into account in the prior distributions of the model. The model is now being applied to the analysis of operating experience in a case study. Later, it will be benchmarked in an international project.

In addition to reliability modelling, the project has contributed in the analysis of ageing of I&C equipment in a pre-project. The findings of this pre-project indicate that technological obsolescence is a more important form of ageing than the ageing of components. Further, need of more systemic approach for ageing management has been identified.

Note: A special report on the PASSI project in chapter 5.11.

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4.3 Methods of risk analysis (METRI)

Improvements both in methodology and acceptability of probabilistic safety analysis (PSA) have made its uses in decision making more frequent. The role of PSA has been defined clearly in the YVL Guide 2.8 published by the Finnish regulatory body STUK. Decisions, where PSA is used to give safety related information are called risk informed. Among the most difficult problems of the risk informed decision making (RIDM) are balancing between probabilistic and other types of criteria. A part of the problem are the uncertainties that exist both in PSA results and in the underlying models.

PSA is currently the only tool for RIDM that is able to comprehensively collect the evidence and uncertainties in one framework. Consequently, the main goal of the project is to improve PSA methods and licensing practices so that better risk informed decisions can be made. Furthermore, the goal is to develop skills in risk analysis in Finland, assure the competence transfer to the new generation and to contribute to international perspectives of PSA.

Main objectives in 1999–2000

The detailed objectives of METRI may be divided into areas ‘Risk Informed Decision Making And Licensing’ and ‘Risk and Uncertainty Analysis Methods’. The specific goals of ‘Risk Informed Decision Making And Licensing’ have been to develop approaches to decision panel studies on RI-ISI (risk informed in-service inspection), to investigate new methods for passive system reliability assessment and to develop methodology for PSA qualification for decision making. The specific goals of the area ‘Risk and Uncertainty Analysis Methods’ have been to develop approaches for uncertainty analysis of deterministic models; to evaluate appropriate importance measures (IMs) for risk informed decision making and to develop more valid human reliability analysis methods including errors of commission and integrated analysis with the WOPS project.

Main results in 1999–2000

An **expert panel methodology** has been developed for RIDM. The method has been applied in the STUK's RI-ISI (Risk-Informed In-Service Inspection) pilot study (Simola & Pulkkinen 1999), which is a trial to use PSA together with deterministic analyses in making safety related decisions. This requires three things. First, quantitative risk estimates and results of deterministic calculations must be collected. Secondly, a structured decision analytic view on the problem must be formed and balanced combination of expertise from several technical areas must be sought to form a panel. In addition to this, also the impact of related uncertainties must be evaluated. The aim of an expert panel is to achieve a balanced utilisation of information and expertise from several disciplines in decision-making. In the RI-ISI application, the expert panel approach was used to combine the deterministic information on degradation mechanisms and probabilistic information on pipe break consequences. The approach has been utilised, e.g., for the assessment of inspection frequency justification of the diesel back-upped seawater system of Olkiluoto and the emergency feedwater system of Loviisa NPPs. The expert panel served both as a critical review of the preliminary results and as a decision support for the final definition of risk categories of piping. The approach enabled a structured discussion between experts from several disciplines, which was felt very useful. Although no clearly contradicting opinions between the experts arose, the panels were seen important as quality assurance audits. Along with the increased adoption of risk informed principles, the use of decision analytic approaches may be inevitable in RIDM. (A separate presentation in chapter 5.12.)

An important issue in METRI has been forming a view on **uncertainty assessment** including consideration of phenomenological uncertainties (Pulkkinen 2000, Simola 2000). Traditionally, probabilistic safety assessment (PSA) is completed with an uncertainty analysis that concentrates upon aleatory – or statistical – uncertainty in the basic event probabilities. This is done by representing them as probability distributions, by propagating the uncertainties in the PSA model by means of Monte Carlo simulation and by studying the resulting uncertainty in the results. As an extension of this approach, the latest developments of PSA uncertainty analyses also include evaluation of uncertainty of physical and logical models. The treatment of this kind of uncertainties differs from the traditional one, and in some cases it may be difficult to do it in a probabilistic way. The fundamental goal is the evaluation of the value of information from PSA. This is manifested from the decision maker's point of view in table 3.

Table 3. PSA-related decision making situations and uncertainty seen from the decision maker's point of view (Holmberg & Pulkkinen 1999).

Compliance with safety targets according to PSA	**Acceptance of PSA's result**	
	YES	NO (results are too uncertain)
YES	No requirements for further actions	Improve quality of PSA/safety analyses or improve quality of the design
NO	Improve quality of the design	Improve quality of PSA/safety analyses or improve quality of the design

The aspects of incompleteness and model uncertainty shall be regarded in addition to the traditional aims, such as identifying the quantitative impact of PSA model basic event uncertainties on the final results. A good uncertainty analysis enhances the transparency of PSA, i.e., it is possible to see how different assumptions and pieces of evidence are related to the final conclusions (Pulkkinen et al. 2000). In METRI, the starting point has been that PSA should be seen as an organised collection of evidence about the safety of the plant, and the purpose of an uncertainty analysis is to document and clarify the evidence behind the PSA results. Each modelling task of PSA includes uncertainties, the type of which may be task-specific. The uncertainty analysis should aim at a transparent documentation of the assumptions and unclear issues of the analysis. The distinction of various types of uncertainty can be used in a decision making situation in order to identify the most suitable measures for uncertainty reduction and for determining the needs for additional evidence.

This kind of broad qualitative uncertainty analysis serves as a basis for determining the requirements for quantitative uncertainty analyses. The quantitative analyses are selected according to the purposes for which the PSA is applied.

In the project, an analysis framework has been developed for uncertainties including the following topics and a specific form (Pulkkinen et al. 2000):

I *Uncertain issue* (identifier and reference to the PSA model)

II *Text description* (verbal description and type of uncertainty i.e. assumption, model, parametric etc. uncertainty)

III *Uncertainty amount* (small, moderate, considerable) and effect (overestimates risk, underestimates risk or direction not known). One must consider separately the issue itself and its impact on results.

IV *Justification to III and used evaluation method* (qualitative, modelling, quantitative)

V *Other remarks* (links between study parts and uncertain issues, comments, ways to decrease the uncertainties etc.)

The approach has been applied in NKS/SOS-2.1subproject ”uncertainties” (Simola 2000). For example, the work has included assessing physical process uncertainty related to the potential leaks of hydrogen to BWR reactor building. The approach has also been applied outside METRI. Recently, a mini seminar was organised on uncertainties of physical processes in METRI. Its presentations will become available, later on, in a specific report.

Passive system reliability and licensing requirements are one application area of the analysis methods for physical uncertainties. The motivation behind studying them is that almost all advanced NPP designs include passive systems dependent on thermal hydraulics or chemistry. However, the reliability analysis and licensing practices of such systems have not been studied sufficiently. This was revealed by an international inquiry in OECD countries carried out as a part of METRI. The work in 1999 included

1) a review of existing approaches to passive system reliability methods, 2) developing classifications for different passive systems and 3) preliminary views for their licensing practices (Pyy 1999a). It is worth noticing that studying physical processes as such may not be sufficient, since the factors disturbing them, such as human actions and changes in environmental conditions, may come from outside a system. The passive system reliability work has continued in 2000 by developing methods for qualitative uncertainty analysis and applying them in an analysis of emergency condenser of a SWR reactor. This application work is ongoing.

An overview on **importance measures** used in reliability engineering has also been structured and reported (Myötyri & Pulkkinen 1999) as a part of METRI. The report describes the properties and definitions of the following importance measures used widely in risk assessments: Birnbaum, Fussel-Vesely, Criticality, Risk Achievement Worth, Risk Reduction Worth and Improvement Potential. In addition to this, the ways to present the results of importance analyses to decision makers are discussed.

The report gives an overview on the definitions, interpretations and the use of the most common risk importance measures. A problem with importance measures is that each of them reflects only certain aspects of the system. In many cases, the IMs may show very different results. Thus, their careless and one-sided use may give a misleading view upon the system and a joint use is, consequently, recommendable. To manifest the need of using several measures, the case of criticality and improvement potential importance measures is shown in figure 25. The figure shows how changes in unavailability affect the relative importance of the basic events.

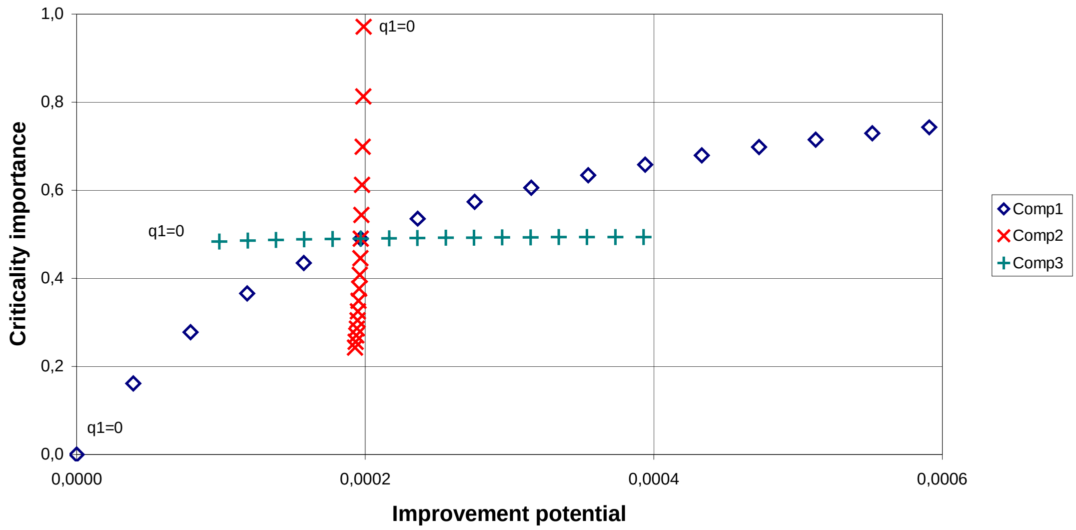


Figure 25. Comparing two importance measures while changing unavailability q_i *of three basic events (comp 1,2,3) in a PSA model. As seen, the relative order of the basic events changes as a function of* qi.

The importance measures not covered in detail in (Myötyri & Pulkkinen, 1999) were uncertainty importance measures, structural importance measures and importance measures for groups of components. It is evident that, for example, structural importance measures would be useful in safety regulation work. Theoretically interesting area are the importance measures related to decision-theory and uncertainty, which is also an important and not well understood issue of risk informed decision making. In 1999, a mini-seminar was also organised on IMs as a part of METRI. The work is currently ongoing in a M. Sc. thesis.

In the field of **human reliability analysis (HRA)**, an EU concerted action (Liwång et al. 1999) has been finalised in the area of integrated sequence analysis and two articles (Holmberg et al. 1999, Pyy 2000a) have been published in a scientific periodical. The project reviewed both dynamic methods in HRA and interdisciplinary approaches. Decision analytic point of view was taken in the Finnish contribution (e.g. Pyy 2000c, d). One of the findings was that several types of different information and competence has to be available for a good HRA. This is depicted in figure 26.

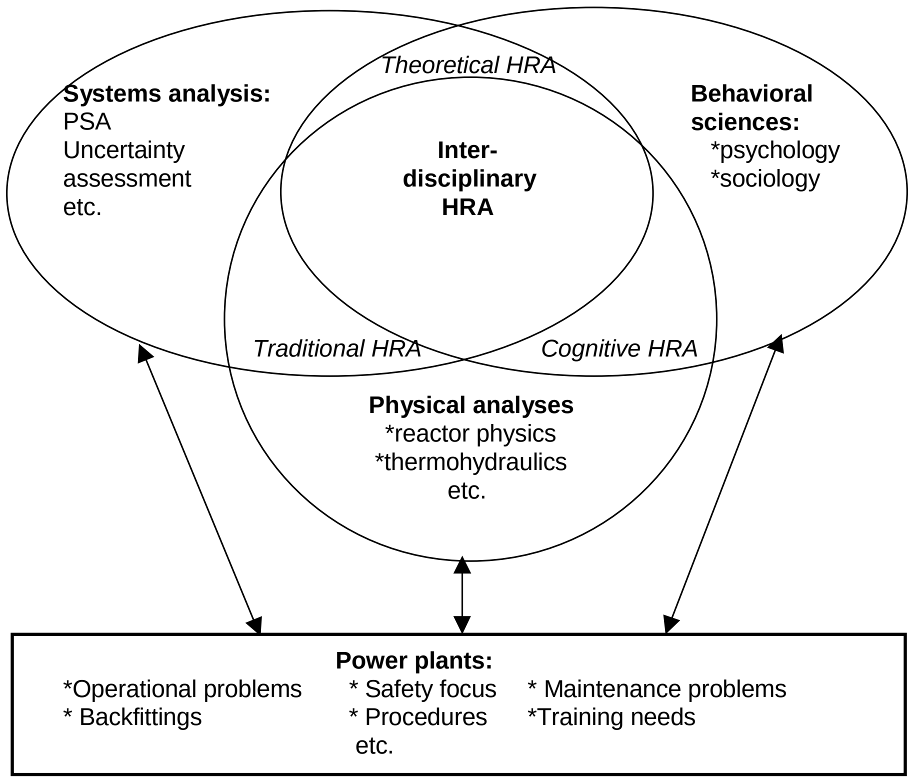


Figure 26. A presentation of the structure of an interdisciplinary HRA modified from Liwång et al. (1999).

The integrated assessment has continued in Finland in the field of fire PSA. The idea has been to investigate if specific HRA methods are required for fire situations. Co-operation has been established with projects FINNUS/WOPS and FINNUS/FISRE. As a part of this work, interviews on human reliability in fire situations have been carried out in the regulatory body, research body and power plant organisations. A specific presentation is given about this issue under project WOPS (chapter 5.13).

Apart from integrated approaches, a framework for identification and probabilistic analysis of human errors of commission (EoCs), i.e. wrong human actions, has been created. The approach has been applied to analysing a disturbance at Loviisa NPP. In the development work, it has been noticed that a division between the omitted actions i.e. errors of omission (EoOs) and the EoCs is not enough in the analysis work. One also has to investigate the effects in the target system and process. For example, there is no mechanism that would lead from an EoO to component unavailability and from an EoC to wrong system response – the system response is completely dependent on the context. This idea, verified by many Finnish studies (e.g. Pyy 1999b, 2000b), is depicted in figure 27.

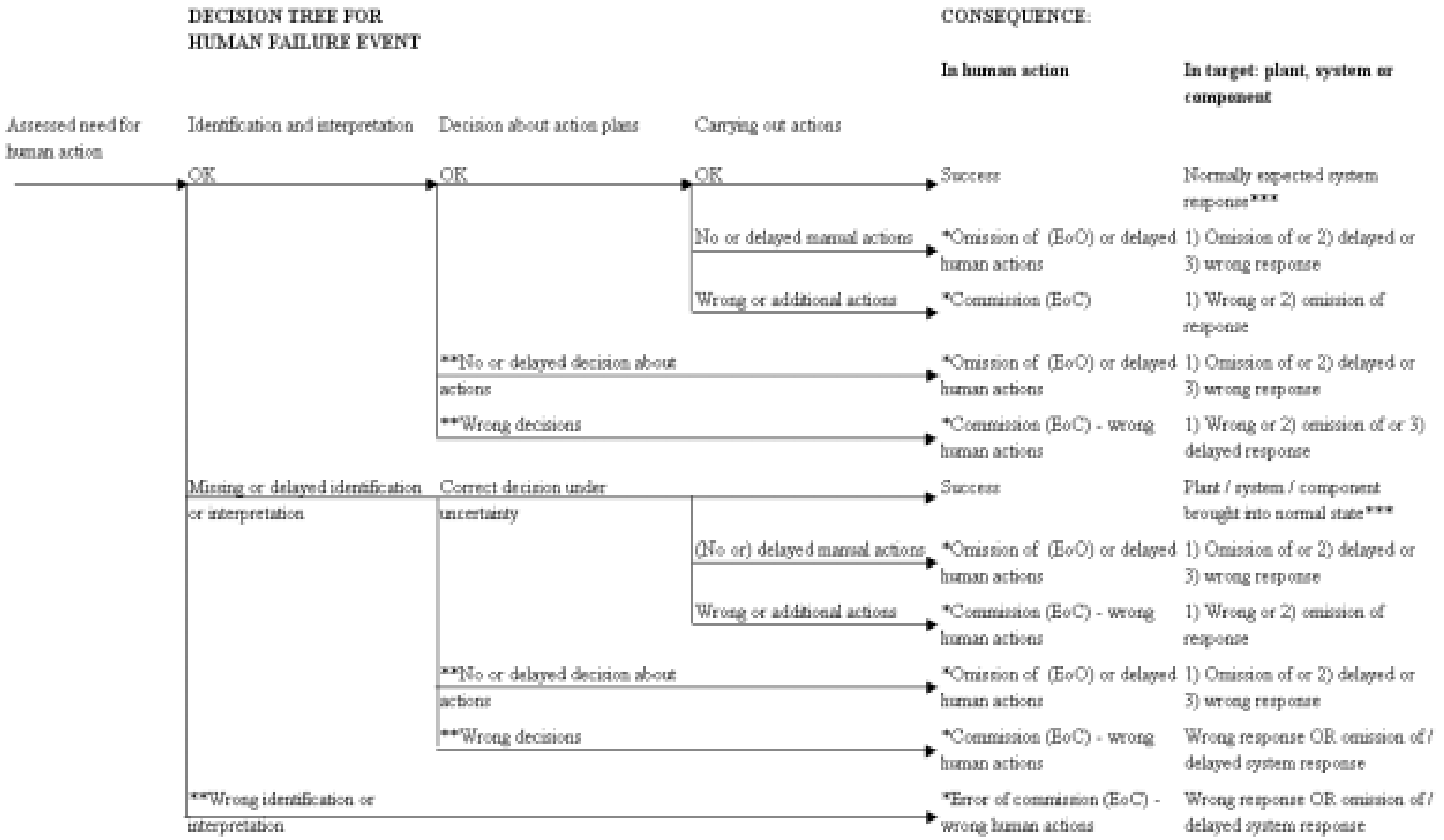
On the international front, METRI has contributed successfully to OECD, NKS and ESReDA co-operation e.g. by organising workshops. A Dr. tech. thesis has been published on reliability methods in ageing management (Simola 1999). More academic theses are currently prepared and, thus, the project fulfils its educational role. Another role is to follow-up other projects in the FINNUS programme and bring available the systems analysis point of view.

Main objectives in 2001–2002

The objectives of METRI for 2001–2002 are to further enhance uses of PSA in decision making and to reduce related uncertainties. The work started in 2000 in the field of qualifying PSA to be used in decision making will be continued. Phenomenological uncertainties, passive systems and human reliability remain as important topics of research. Pipe failure frequencies will become a new theme and they will be investigated from RIDM point of view. As a specific objective, academic degrees and scientific articles are continuously aimed at.

Applications

The results of METRI have direct applications in risk informed decisions both in the utilities and in the regulatory bodies. Examples of such decisions are planning of maintenance, inspection and plant backfitting programmes. Through a more formalised decision panel approach, decisions and the reasoning behind them become more transparent to all interest groups. Uncertainties are always present in decision making.



* In all cases, identification or decisions may be corrected given that the target system and time allow (recovery)

** No double failures included, e.g. no wrong identification and additional delayed manual actions

***Given that latent failure mechanisms are not present in the man machine system.

Figure 27. A decision tree depicting the deviations in different phases of human actions and their consequences both in human performance and in target system.

METRI has applications also in bringing new ideas in classifying different types of uncertainties and comparing their type, size, effect and ways to reduce them. Also phenomenological uncertainties and human actions are included in the analysis. Consequently decision makers have a better view on different aspects and uncertainties affecting their decisions. Apart from this, the project contributes to the international scientific discussion in its area by regularly publishing articles. Educating own staff and interest groups on probabilistic and decision analysis issues is another field of activity.

Note: A special report on the METRI project in chapter 5.12.

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4.4 Working practices and safety culture in nuclear power plant operations (WOPS)

The aim of the project is to enhance safe operation and maintenance of NPP. Based on analysis of human decision- making and action in actual working situations, practical methods are constructed for the evaluation and development of working practices and safety culture in the plants. Human factors analysis and advanced risk analysis are integrated to provide better methods for the evaluation of the overall safety of the plants. A conceptual framework for the analysis of activities in dynamic environments has been developed starting from the notion of human activity as intentional, context-dependent and situationally constructed enterprise. The methodology is called contextual analysis of working practices (CAWP). The ongoing work in WOPS aims at further development of the method.

Main objectives in 1999–2000

The main objectives were to analyse human reliability in fire situations, organizational and safety culture in the Radiation and Nuclear Safety Authority of Finland (STUK), coping with demanding operation situations and human factors in maintenance. In addition a study concerning change of NPP employee generation and their socialisation and development of expert identity was started. The study focuses on operator trainees.

A further aim was participation in the international co-operation both in the NPP area (NKS SOS 1 and 2.2 and OECD/NEAPWG 1) and EU thematic network of Work Process Knowledge (WHOLE). In NKS co-operation the aim was a benchmark safety culture study in Sweden and Finland.

The main methodological objective of the project was to broaden out the methodology to take into account 1) cultural and organisational aspects 2) emotional and energetic aspects of human performance and learning 3) develop realistic human reliability methods integrating psychological approaches with probabilistic safety analysis. The further challenge was to integrate the applied and developed theories and methods to the previous studies and the theoretical background of CAWP.

Main results in 1999–2000

Theoretical and methodological work has been proceeded in several themes: the overall method for work analysis, cultural and organizational aspect of work and emotional-energetic dynamics of performance and learning in work. Based on the theoretical work and empirical studies new concepts and the first versions of methods have been developed. Each of these is handled below in subchapters.

The core task analysis has been developed further (Norros & Nuutinen 1999, Reiman & Norros 2000, Norros & Nuutinen in preparation). The concept of *core task* refers to essential content of particular work. The goal is a methodology for the analysis and development of expertise in process control work. Core task analysis is seen as a tool to reflect current practices and develop new ones. The needs for reflection are basically two. The first relates with the difficulty in normal practice to identify the essential functional demands of work and to maintain orientation to them also when their critical role is not evident. The other motive for reflection is that there are continuous pressures for change in the activity system, which demand re-evaluation and re-definition of the content of work and the core task.

Human reliability in fire situations. An integrated approach concerning human reliability in NPP fire situations has been developed in co-operation with the METRI project. Relevant literature has been considered focusing on the way the special features of fire situations have been taken into account in the related human reliability analysis (HRA). In order to gain a comprehensive view of the nature of NPP fire situations, expert interviews have been carried out. Part of the experts represented the deterministic and probabilistic safety analysis in nuclear power plants, STUK and VTT. Other experts were simulator trainers, control room operators and members of the fire departments of the plants. The aim of the interviews was to find out problems of fire fighting and simultaneous process control, and, in addition, problems of related deterministic and probabilistic research. On the basis of the literature and the interviews, a method has been developed for the conceptualisation of the cognitive demands of process control in NPP fire situations. It describes fire situations from the operators' decision making point of view, focusing on the role of the dispersed and possibly reduced and misleading information. The purpose of the analysis is to offer a complementary view for the existing HRA methodologies and expert judgement methods. The results of the study can be utilised also in the development of training and procedures, both on the fire fighting and the process control side. (See WOPS special report in this volume).

In **Organizational culture and safety culture research** the ”safety culture” concept has been specified (Reiman 1999) and the first version for a methodology to study safety culture within high reliability organisations has been proposed (Reiman & Norros 2000).

A case study was conducted at STUK's Nuclear Reactor Regulation (YTO) -department. Objectives of the study were to conceptualize and describe YTO's organizational culture and define its core mission and the requirements that the mission sets for inspectors. We used a combination of quantitative and qualitative methods in the research. Preliminary investigations consisted of document analysis and interviews. First phase of the project consisted of a survey method (organizational culture survey and job motivation and stress survey) and second phase consisted of a workshop for the whole staff. Results were analyzed from the viewpoint of the organizational culture and from the viewpoint of the YTO's core mission. Further development and assessment of the used method was carried out after the case study concluded.

The results of the case study show that the employees at YTO value professional knowledge, openness, courage, fairness, efficiency, questioning attitude, teamwork and independence. These values can be interpreted as a YTO's conception of an ideal regulatory culture.

When discussing targets for development social aspects of the work became central: social and professional support of co-workers, socialization of newcomers (transfer of knowledge and experience), internal communication and internalization of YTO's values. Also resource and work-process management were mentioned as targets for development. A wish for the clarification of the YTO's goals and the enhancement of feedback at both individual and organizational level were emphasized in both the survey and workshop results. When discussing potential future threats to YTO's culture, too heavy bureaucratization and the subsequent loss of a meaning of one's work were mentioned.

Regulator has an indirect influence on the safety (and safety culture) of the power plants. Because the influence is indirect, it is hard to see the results of one's work. This means that the feedback from one's actions is also indirect. Usually external feedback is mainly negative. But at the same time employees acknowledge the importance of their job. This can cause stress and burnout. The results point out to the importance on internal feedback from managers and colleagues.

The core task analysis of the research material also show that the regulatory culture is composed of three subroles, the authority role, the expert role and the public role. These roles set different demands for the regulatory culture. The role of the expert demands flexibility and creativity (creation of new knowledge). The public role demands openness and communication. The role of the authority demands control, measurement, documentation and processing of information. Professional knowledge is a highly emphasized value in YTO's culture, but its practical implications are not so clear. Regulator needs ways to maintain, develop, transfer and capitalize its expertise.

The paper presented in the Workshop on New Technology and Work in Bad Homburg (Reiman & Norros 2000) was the first step to integrate the theoretical background of the method developed in the organizational culture study and the previous CAWP methodology.

In **international co-operation** the safety culture study at all Finnish and Swedish nuclear power plants was completed. A similar study targeting on quality assurance, quality systems and quality thinking has been initiated. The first interviews of this study has been carried out and the study is expected to be reported early year 2001. A follow up of the EU funded ORFA project is under preparation.

Coping with demanding situations. More attention was paid to the subjective and emotional aspects of behaviour in the situational organisation of actions, in particularly regarding the mobilisation of own energetic resources. The result was *a motivational performance model (MPM).* The model offers a conceptual framework to handle questions like NPP personnel's motivation and commitment contextually and together with their job demands.

An expert identity concept is central in the model. This concept tries particularly to capture the emotional aspects intertwined with work performance, especially coping with demanding situations, and development of expertise in work. According to Lave (Lave & Wenger 1991) the development of identity and knowledge and skills are part of the same process. In practical action both individual and societies shape themselves and each other. In this process the effort to develop identity serves developing skills by giving motivation, formation and meaning. The purpose of development is a master identity, which gives the full membership to community. The problem with the change of working environment is that it breaks up the control/master identity. The change moves one towards the place of a beginner.

The expert identity is defined to consist of three components:
1) Meaningfulness, the sense of the importance of persons' own work
2) Professional self-confidence
3) Sense of control.

These three components are inter-related and affect each other. Meaningfulness refers to the person's feeling of the general and personal importance of her/his profession or work. This component could be evaluated against a core task. In an ideal situation the sense of meaningfulness is based on aspects of the core task of work according the model. Then it is in line with the goals of the nuclear power production (e.g. safety and productivity) and contextual and situational constrains of work (e.g. information and automation systems of process control). Self-confidence of an expert is related to the feeling of possessing skills, knowledge and experience, by which she/he believes she/he can take care of the work situations. This confidence refers also to co-operation. A hypothesis is that strong self-confidence based on realistic understanding of one's expertise makes an effective co-operation possible, because the person feels not to be in a danger when she/he is in co-operation with others. The sense of control component is a result of the interaction between the situational control of one's actions and the emotions awoken by reaching goals. Thus, a positive emotion (exercising control) energises actions and focuses it on target. A negative emotion (loosing control) interrupts action and makes possible to re-direct action, but it can also paralyse the action if the sense of control is totally lost. Then the person can try to find the sense of control with another goal, for example "saving her/his face" and the action is directed to the new goal. So the goals can be in line with, for example, safety directed actions or self-protection. The importance of the emotions in controlling one's action is based on Oately's and Johnson-Laird's Communicative emotion theory (Oatley 1992).

HF in NPP maintenance. Prerequisites for human factors studies in maintenance has been created by participating in a maintenance study. The expert judgement method was described and reflected.

Main objectives in 2001–2002

The main objectives are to complete the study concerning change of NPP employee generation and their socialisation and development of expert identity and analyse and development of human reliability in maintenance. A further important aim is to crystallize findings, theories and methods by publishing in national and international journals.

Applications

Developed and developing practical methods based on theoretical and empirical work apply for the evaluation and development of working practices and safety culture in high reliability organizations. Integrated human factors analysis and advanced risk analysis is going to provide better methods for the evaluation of the overall safety of the plants. The results of the human reliability in fire situations study can be utilised also in the development of training and procedures, both on the fire fighting and the process control side.

Note: A special report on the WOPS project in chapter 5.13.

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5. Special reports in the Interim Seminar 31.10.–1.11.2000

5.1. AGE special report

Activity build-up and corrosion

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Abstract

Due to the exposure to the aqueous coolant, the components of the primary circuit in a nuclear power plant oxidise, resulting both in the formation of an oxide film on the component surfaces and in the release of soluble metal ions. The properties of the oxide films are likely to have a significant effect on the corrosion susceptibility as well as on the rate at which radioactive nuclei are incorporated on the component surfaces from the coolant. The latter phenomenon has a marked effect on the occupational dose rates, especially during annual outages. This work aims at understanding the properties and behaviour of oxide films in different coolant conditions and thus at predicting the risks for corrosion and activity build-up. The approach chosen is to investigate and model separately the phenomena taking place at the film/solution interface, in the outer, porous part of the oxide film and in the inner, compact part of the film. In addition, a significant contribution is paid to the development of electrochemical techniques capable of operating in high-temperature environments with even extremely low conductivity. This report summarises the progress made during the first two years of the project.

Introduction

This work aims at predicting the influence of operational conditions of nuclear power plants on occupational dose rates and on the corrosion susceptibility of the plant components when exposed to coolant water. This will facilitate assessing the impacts of ageing, of introduction of modified water chemistries and of increased power output in Finnish power plants.

Occupational dose rates especially during maintenance periods depend on the extent of incorporation of radioactive species from the coolant on the primary circuit components. Increased incorporation rates may lead to a need to decontaminate the plant components, as has been done successfully at Loviisa 2 power plant in 1994. To assess and influence the incorporation rate of radioactive species on the component surfaces, it is essential to understand the mechanism and to identify the rate-limiting steps of their incorporation. This in turn requires modelling of the processes taking place on the component surfaces.

The surfaces are covered with oxide films due to the exposure to the oxidising coolant water. The oxide films usually comprise different layers with different physical and chemical properties. The scheme of an oxide film forming on stainless steel in typical BWR conditions (figure 28) demonstrates that the outer part of a steady-state oxide film is porous, while the inner part is more compact. The films forming in PWR conditions are closely similar to that shown in figure 28. The incorporation of radioactive species into the oxide may thus be controlled by surface phenomena and by the transport or species in the outer layer or in the inner layer. The incorporation rate is also necessarily influenced by the nature of the incorporating species. Competitive incorporation of less harmful species may reduce the rate of activity incorporation, which pinpoints the important role of the coolant composition as well. A novel method that has been found to lead to a decreased rate of activity incorporation is injection of zinc into the primary coolant. In order to optimise the injection procedures of zinc and to find possible alternatives, it is essential to identify how it influences the different stages of incorporation.

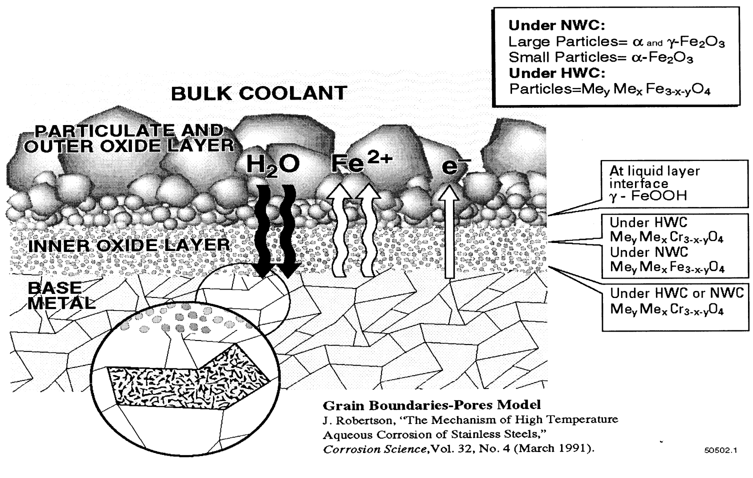


Figure 28. A scheme of an oxide film forming on stainless steel in typical BWR conditions (Baston et al. 1996).

The compact, inner oxide layer may not only influence the rate of activity incorporation but it is also likely to determine the rate of the corrosion of the component material. This fact becomes especially important when the whole oxide film is thin and probably consists of the compact part only, which may be the case due to some local damage to a steady-state film. In such cases the protective properties of the compact part may be strongly connected with the susceptibility to environmentally assisted stress corrosion cracking (EASCC) and other forms of corrosion.

Sufficient experimental data and reliable models concerning the behaviour of surface oxide films in conditions corresponding to NPP coolants are not available. Fortunately, the experimental techniques developed in the RAVA5 project during 1997–1998 (Laitinen et al. 1998) have made it possible to obtain more in-situ information of the properties of oxide films forming on material surfaces in high-temperature aqueous environments. In addition, models to describe the adsorption phenomena on oxide surfaces and possible processes in both the porous and the compact part of the oxide film at room temperature are continuously developed. These developments have made possible to start the present work in which the processes in different parts of the oxide film influencing the activity incorporation and corrosion are studied and modelled in high-temperature conditions. As a starting point, the film is assumed to remain in a steady state, which means that its thickness and composition do not change significantly with time. Thus, transient situations in nuclear power plants are not considered in this modelling work, but they will be a subject for further investigations.

Adsorption of radioactive species from the coolant

This task aims at:

* understanding how radioactive species such as Co-60 behave when they reach the outer oxide surface
* which factors influence their adsorption
* how these factors are affected by prevailing conditions such as coolant composition.

While considering the competitive interaction of various dissolved ions with the oxide film, we consider the first step to be adsorption. It is for instance possible that the influence of zinc on Co-60 incorporation is due to the competition between zinc and cobalt for the adsorption sites, as discussed by Mäkelä et al. (1999) in a recent literature survey.

In this work, the surface complexation approach (Stén et al. 2000) is taken to elucidate the effects of solution conditions on adsorption. In contrast to the classical adsorption models, surface complexation models represent adsorption in terms of interaction of the adsorbate with the surface OH groups of the adsorbent oxide resulting in the formation of specific surface complexes. Despite the uncertainty in the exact composition and structure of the iron oxide film formed on the iron and steels, it can with good reason be assumed that the hydrated passive film on iron and steels displays the coordinative properties of the Fe(lll) surface hydroxyl groups, ≡Fe–OH. These groups are ampholytes being able to react as follows:

$$\equiv Fe\text{–}OH + H^+ <=> \equiv Fe\text{–}OH_2^+ \quad (6)$$

$$\equiv Fe\text{–}OH <=> \equiv Fe\text{–}O^- + H^+ \quad (7)$$

Metal adsorption on the deprotonated surface hydroxyls can be described by various surface complex formation reactions, e.g.

$$\equiv FeOH + Co^{2+} <=> \equiv FeOCo^+ + H^+ \quad (8)$$

$$\equiv FeOH + Zn^{2+} <=> \equiv FeOZn^+ + H^+ \quad (9)$$

Model calculations have demonstrated that the retarding effect of zinc on the adsorption of cobalt on iron oxide can be theoretically predicted on the basis of the surface complexation approach. This has been shown for conditions resembling those prevailing in BWR coolants, except for the temperature which has been taken to be 25°C for reasons explained below. Simulated cobalt adsorption edges in the presence of various amounts of zinc are shown in figure 29. The total cobalt concentration (i.e., the sum of dissolved and adsorbed amounts of cobalt divided by the solution volume) in these simulations is taken as $5 \cdot 10^{-9}$ M (0.3 ppb). The values on the *y* axis give the adsorbed fraction of cobalt as a function of solution pH. Changing the total zinc concentration from $1 \cdot 10^{-10}$ M (0.007 ppb) to $1 \cdot 10^{-7}$ M (7 ppb) already leads to a situation in which the more strongly binding zinc effectively retards cobalt adsorption considerably. The latter is in the range of Zn concentrations in the reactor water of plants which are dosing Zn (Mäkelä et al. 1999). At pH 7.6 (indicated by the vertical line) the addition of $1 \cdot 10^{-5}$ M (700 ppb) zinc results in the reduction of cobalt adsorption from 80% to 20% on the oxide surface.

As the general trend is, that an increase in the temperature of the system promotes significantly cation uptake (Cornell & Schwertmann), we expect that at elevated temperatures similar results are obtained at lower pH values. However, quantitative modelling of adsorption phenomena at elevated temperatures in terms of the surface complexation models requires information on the temperature dependence of the equilibrium constants of the surface and solution reactions. Finding and assessing this data along with experimental adsorption studies will be one of the main tasks during the rest of this subproject. This allows to predict adsorption behaviour in relevant conditions.

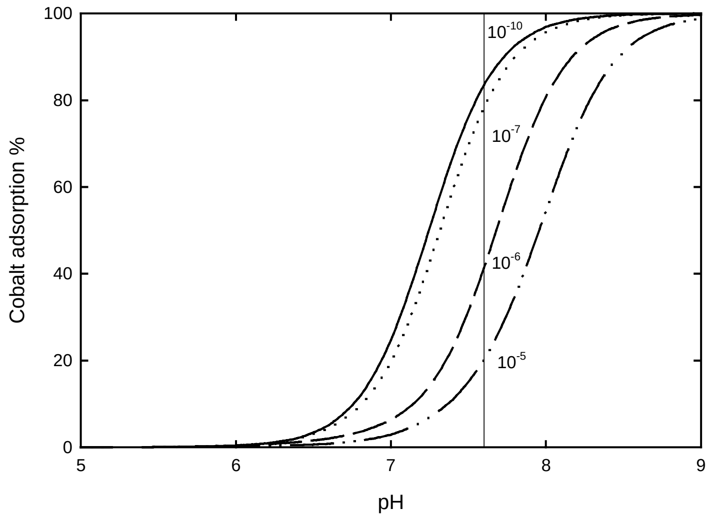
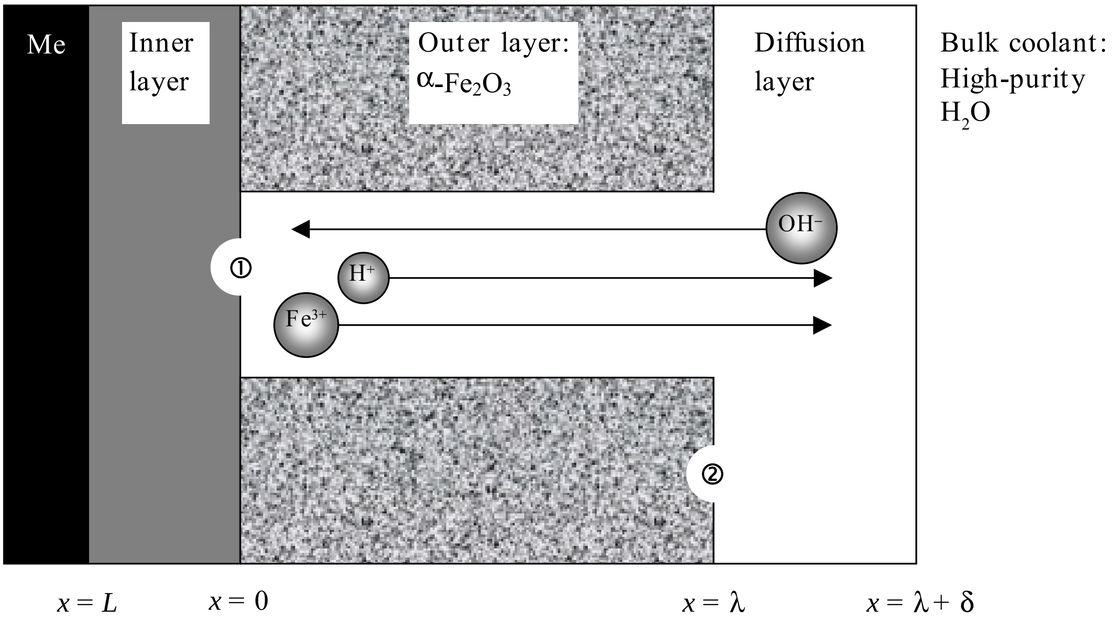


Figure 29. The effect of increased zinc concentration (from 10^{-10} M to 10^{-5} M, i.e. from 0.007 ppb to 700 ppb) on the adsorption of cobalt on iron oxide from water containing $5{\cdot}10^{-9}$ M (i.e. 0.3 ppb) cobalt. A typical Zn content in the reactor water in the plants dosing Zn is 1 to 10 ppb.

Role of the outer oxide layer in activity incorporation

This task aims at modelling steady-state transport phenomena in the deposited, porous oxide film corresponding as closely as possible to the one met in coolant systems in nuclear power plants. The motivation for this is that the outer oxide layer most probably influences the steady-state incorporation rate of radioactive species by means of determining their rate of transport towards the interface between the outer and inner layer. The thickness of this outer layer in typical simulated BWR conditions is a few hundred nm (Lister & Venkateswaran 1999) while its thickness in typical simulated PWR conditions is smaller (Da Cunha Belo et al. 1998). In real plant conditions the oxide film thicknesses may be considerably higher (Aaltonen et al. 1995).

The work done so far (for more details see for instance Lehikoinen and Olin, 2000) consists of completing a first version of the computational model, which accounts for the transport of molecular species. Thus it can also be applied to the incorporation of radioactive species. A BWR-type coolant has been selected as a modelling medium. It also serves as a template for further model development. A plausible set of governing equations has been solved for the movement of species within the deposited oxide film subject to both concentration and electrical potential gradients in high-T water. A scheme of the chosen approach is shown in figure 30. In this first approach, the thickness of the outer, porous part has been assumed to be of the order of 400 nm.



Model SAT

① Film production at $x = L$ and $x = 0$:

$$2Fe + 3OH^- \rightarrow Fe_2O_3 + 3H^+ + 6e^-$$

② Film dissolution at $x = \lambda$:

$$Fe_2O_3 + 6H^+ \rightarrow 2Fe^{3+} + 3H_2O$$

Figure 30. A scheme of the concepts for modelling the transport of molecular species within the porous layer of the oxide film on primary circuit surfaces. Model SAT, i.e. saturation with respect to hematite α-Fe_2O_3.

The results have made it possible to estimate, for instance, the potential drop within the pores of the layer as a function of different parameters. The potential drop contributes to the establishment of a possible driving force for the transport of radioactive species within the porous layer. The preliminary results shown in figure 31 indicate that the absolute values of the potential drop established along the pores are relatively low. The porosity of the film can be seen to have a strong effect on the potential drop. The influence of this drop on the transport of species in the pores can be more quantitatively assessed after further development of the model.

The next step in the modelling of the porous layer comprises the extension of the model to account for common aqueous and surface composition ranges of high temperature systems in nuclear power plants and also for the incorporation of radioactive constituents on the oxide phase by way of surface complexation.

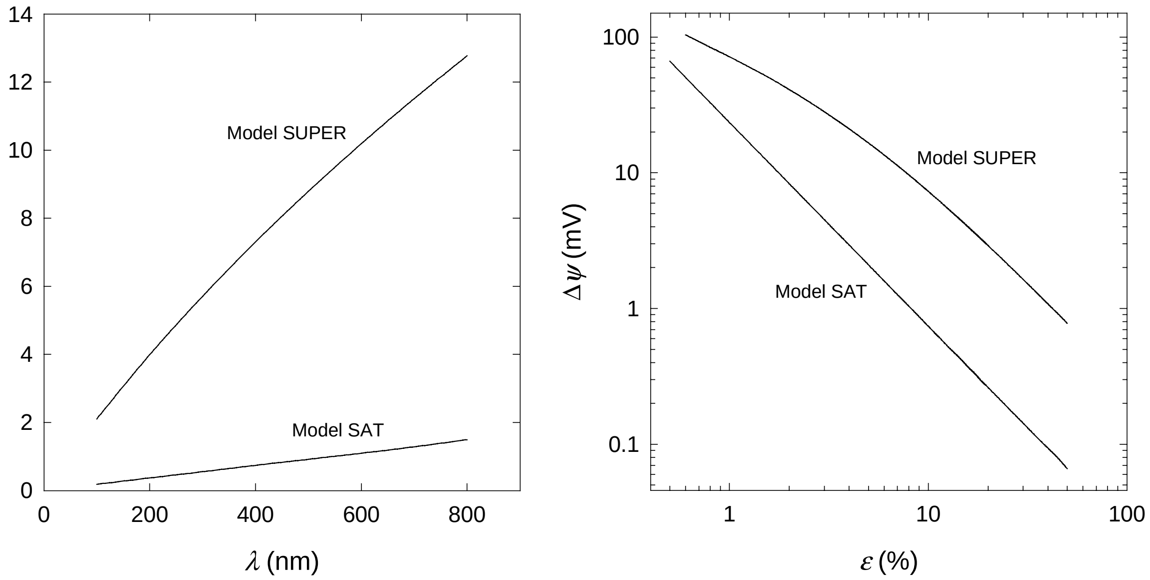


Figure 31. Variation of the potential drop with the thickness (λ, left) and porosity (ε, right) of the deposited oxide (hematite) film for Models SUPER (supersaturation with respect to hematite) and SAT.

Role of the compact, inner oxide layer in corrosion and activity incorporation

The aim of this task is to model the reactions and properties of the compact part of the oxide film in such a way that the effects of material composition and of the environment on corrosion can be quantitatively accounted for. As discussed above, the compact, inner part of the oxide film influences the rate of oxidation of the component materials and has thus a great effect on different corrosion phenomena and most probably also on the activity incorporation. The thickness of this compact part in simulated conditions may reach values in the order of magnitude of 100 nm (Da Cunha Belo et al. 1998), but it may become markedly thicker in real plant conditions (Aaltonen et al. 1995).

The oxide films forming on material surfaces contain a significant number of ionic defects, the nature and amount of which influence both the properties of the film and the transport of species through it. The presence of ionic defects offers routes for ionic species to be transported through the film, making both the incorporation of radioactive species and the corrosion of metal through the film possible. It can be generally stated that the more defects the film contains and the higher their mobility is, the more susceptible the underlying metal is to corrosion. The structure of the film may also determine which kind of radioactive species are able to enter the film structure or to be transported through the film.

The mixed-conduction model (MCM) introduced recently for room-temperature films (Bojinov et al. 2000a) makes it possible to explain qualitatively the effect of corrosion potential (in other words, the amount of oxygen or other oxidising agents) and of the film structure on the distribution of ionic defects in the film. This is demonstrated schematically in figure 32.

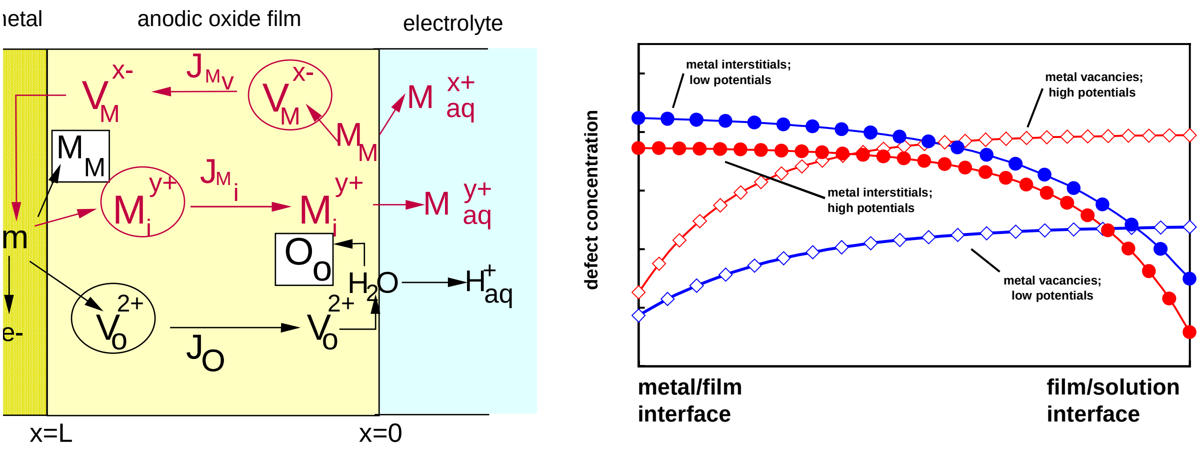


Figure 32. A scheme used in the mixed-conduction model (MCM) of the reactions and transport of ionic defects in an oxide film on a metal surface (left). Distribution of different ionic defects within the film at different potentials. The calculated results are based on the mixed-conduction model (MCM) (right).

The application of the concepts of the MCM to films formed on Fe-Cr and Fe-Cr-Mo model alloys, pure Fe and pure Cr as reference materials and on commercial alloys is in progress (Bojinov et al. 1998, Bojinov et al. 1999a, Bojinov et al. 1999b, Bojinov et al. 2000b, Bojinov et al. 2000c, Bojinov et al. 2000d). The fitting of the model to experimental results obtained at room temperature and at 200°C (Beverskog et al. 2000a & b) suggests the interpretations shown in figure 33. The results indicate that the films on pure Fe grow considerably in thickness as the temperature increases from room temperature to 200°C, while the film on pure Cr seems to be almost unaffected. The behaviour of Fe-Cr alloys is intermediate between pure Fe and pure Cr. When the film on Fe-Cr alloys is depleted in Cr at high potentials, it starts to grow, resembling the behaviour of pure Fe.

The next step in the study of compact oxide films is to perform measurements in a wider temperature range and to determine the differences in film behaviour in a more quantitative way. This requires ex-situ determination of film thickness. Thus one of the most important tasks is to find out whether SIMS, ESCA, AFM, microscopy or a combination of these is the most reliable technique for this kind of determination.

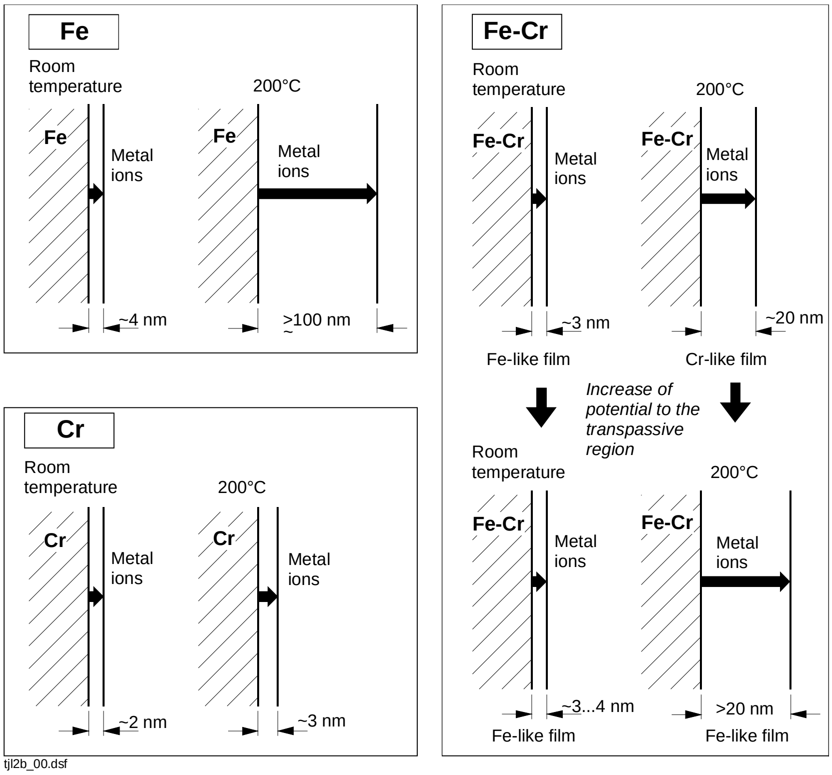


Figure 33. A scheme of the influence of temperature, material compositions and potential on the oxide films forming on metal surfaces.

Electrochemical techniques for high-temperature measurements

The experimental data used as input for the modelling work carried out in this project has required the development of new electrochemical techniques. Conventional electrochemical arrangements are usually not suitable for the high-temperature, high-pressure environments and poorly conductive media encountered in NPP cooling systems.

The scheme given in figure 34 depicts the developed controlled-distance electrochemistry (CDE) arrangement (Bojinov et al. 2000e). When the gap between the working and reference electrodes is large, the system can be used conventional electrochemical measurements. However, by decreasing the gap, the following measurements have become possible:

Thin-layer electrochemistry (TLEC) measurements

* Thin-layer electrochemical impedance measurements to characterise the oxidation and reduction kinetics and mechanisms of metals as well as the properties of metal oxide films even in low-conductivity aqueous environments.

* Other controlled potential and controlled current measurements in low-conductivity aqueous environments.

Wall-jet measurements

* Detection of soluble species released from the working electrode (wall-jet ring-disc).

* Influence of flow rate on electrode reactions.

Contact measurements

* Contact electric resistance (CER) technique (Saario et al. 1998) to investigate and to monitor the electronic properties of surface films.

* Contact electric impedance (CEI) measurements (Bojinov et al. 2000f) to measure he solid contact impedance spectra of oxide films.

The CDE arrangement is currently being applied also to study the corrosion rate and mechanism of fuel cladding materials. The next step in the development of facilities for high-temperature in-situ measurements is to improve the electrical insulation of the electrodes and to optimise the flow conditions in the wall-jet ring-disc configuration. In addition, the assembling of a photolectrochemical set-up for the identification of oxide films at high temperatures and the testing of an in-core Pd reference electrode are in progress.

Conclusions

The main conclusions from this project at this stage can be summarised as follows:

- The surface complexation approach can be used to assess the adsorption behaviour of radioactive nuclei from the coolant on the primary circuit surfaces. However, a quantitative treatment of this contribution to activity build-up requires numerous high-temperature thermodynamic data that is not yet easily accessible.

- The modelling of the transport in the porous part of the oxide film has made it possible to assess the factors influencing transport of ionic species through the pores of the film in different coolant conditions, depending on the film properties.

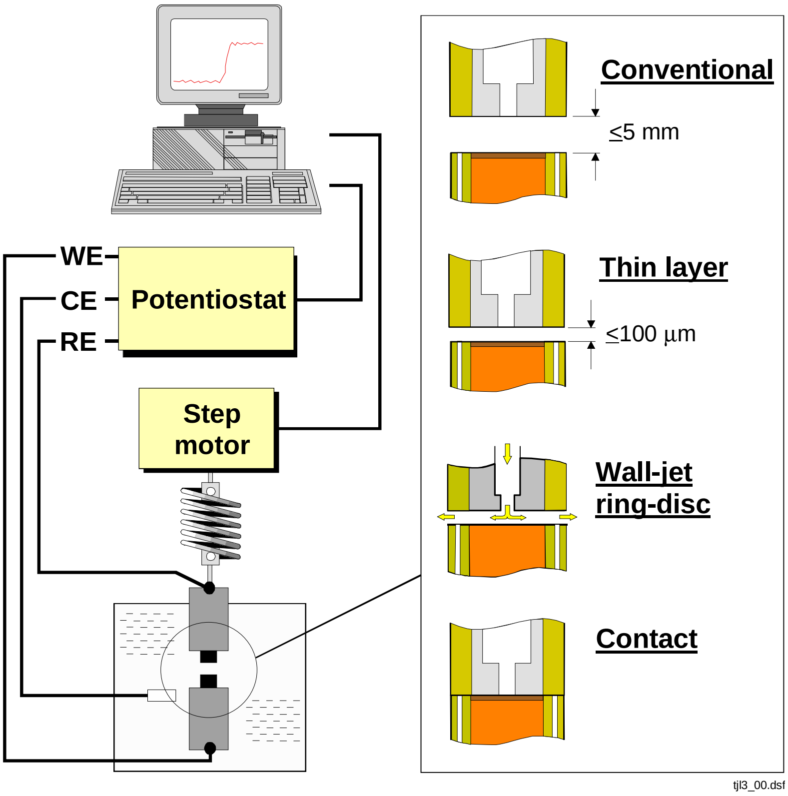


Figure 34. A scheme of the controlled-distance electrochemistry (CDE) arrangement and its various applications.

- The application of the mixed-conduction model (MCM) to electrochemical data obtained in high-temperature environments has proven to be successful. The results indicate that an increase of temperature has the larger influence on material behaviour the larger the Fe content of the material is. A high Fe content results in a high transport rate of species through the film, which in turn results in high corrosion rates and a poorer protectivity of the film.

- The development of a controlled-distance electrochemistry (CDE) arrangement makes it possible to perform versatile electrochemical experiments and to introduce even new techniques in high-temperature aqueous environments.

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5.2 STIN special report 1

Loading to structures due to thermal stratification in a T-joint of hot and cold pipes

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Abstract

Thermal stratification of hot and cold water may cause damages in pipes of power plants. The main aims of this study are to improve the knowledge of the effects of thermal loads on the structures and to develop and verify a computational tool for numerical simulation of structural behaviour under a thermal stratification.

A well-instrumented T-joint test section of hot and cold pipe is used for the experiments. In the hot horizontal pipe, the flow velocity is high, and therefore the flow is strongly turbulent. In the vertical cold pipe, the flow velocity is very small and the flow is laminar. Temperature, strain and displacement measurements will be carried out in order to get data to verify the numerical results.

Preliminary results of the computational fluid dynamics (CFD) calculations are presented. Both the flow equations and the thermal conduction in the pipe walls and insulators are solved with the Fluent code. Transition from turbulent to laminar flow occurs near the T-joint in the cold pipe.

The development of the tools for transferring the data of the CFD calculations to the input needed for structural analysis is under way. Preliminary structural analysis was carried out using the surface mesh of the CFD-model and idealised boundary conditions. The temperature distribution calculated by the CFD-code was transferred to the input file of the finite element (FE)-code. Thermal stresses and strains were calculated by the ABAQUS code.

The numerical results are discussed and they will be used for the further planning of the tests.

Introduction

Thermal stratification takes place in different systems and components in power plants. Moving or cycling stratification layer may cause thermal loads to the structures. The original reason for cyclic behaviour may be thermal expansion of the structures, leaking valve, vibration of the pipeline, operational transients or turbulence of the main pipe interacting with thermally stratified layer in a branch pipe. The effects of cycling thermal loads appear with ageing of the plants.

Stratification in large diameter pipes has been studied in HDR-experiments, where stratified flow was created by feeding cold water to the main pipeline (Eerikäinen 1999). Stratification can also occur in smaller pipes.

Basically, structural integrity can be predicted quite reliably, if thermal loads and boundary conditions are known. In practise, this is not the case. The structural analysis would be more accurate, if thermal loads to the structures could be predicted by the computational fluid dynamics (CFD) codes.

To improve the knowledge of the effects of thermal loads on the structures caused by thermal stratification, the stratification and cycling are studied in TOKE- and STIN-projects in the FINNUS programme. A well-instrumented T-joint pipeline test section is used for the experiments. The test results will be used for verifying the capability of the CFD-codes to predict the flow behaviour in the pipelines. When the CFD-analyses are reliable, the results can be used for the actual structural analysis to predict more accurately the structural integrity. Also, strain and displacement measurements during the test will be carried out in order to verify the results of structural analyses. In this report, the current status of the work for predicting the effects of the thermal loads is reviewed.

Experimental Set-up

The experimental set-up is based on the geometry of the connection line between hot and cold legs in the Loviisa NPP. Cracks near the T-joint caused a leak from the primary circuit (Hytönen 1998). According to the material analysis studies the strong thermal stratification has accelerated the growth of the cracks. In the test loop, a smaller (ID=50 mm) vertical pipe is connected to the bottom of a larger (ID=243 mm) horizontal pipe, see figure 35. Turbulent, hot flow in the larger pipe and cold stagnant fluid in the vertical pipe may cause cyclic temperature behaviour near the connection (Kim et al. 1993).

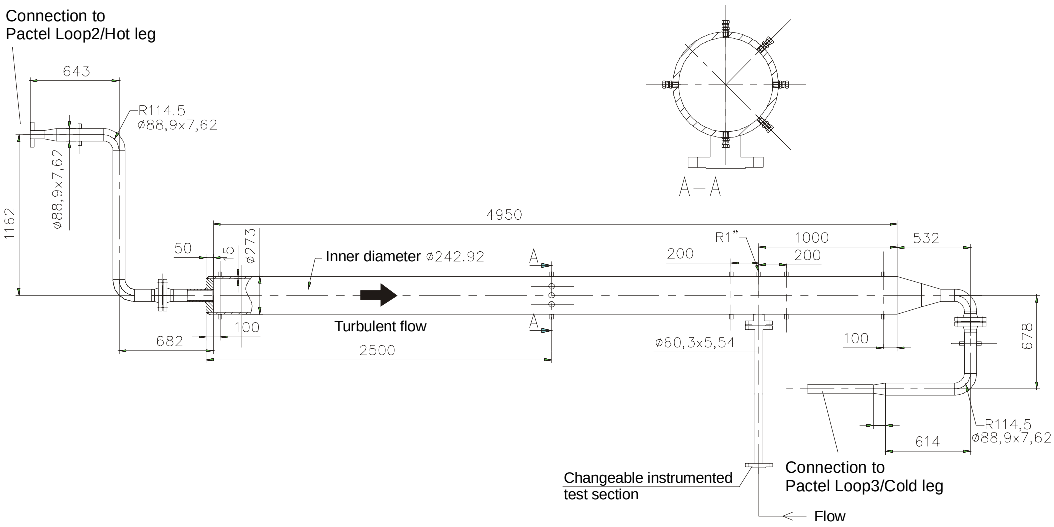

Figure 35. Experimental set-up.

In the first phase of the experiments, the location of the steepest temperature changes will be detected using different flow parameters in the pipes. In the second phase, more detailed instrumentation will be used for determining the temperature behaviour of the fluid and the structures.

The test loop is connected with pipe elements to the nearby PACTEL facility, which is used to produce heat and flow for the tests (Puustinen 2000). The desired temperature range of 250 °C of the flowing fluid can be easily obtained with the use of the 1 MW PACTEL core simulator. The heaters of the PACTEL pressurizer maintain the system pressure at about 7.0 MPa during the tests. The flow is produced with the help of one primary circulation pump of the PACTEL facility. One pump can produce a flow of ~90 m^3/h. In the second phase, two parallel pumps may be used in order to increase mass flow in the horizontal main pipe.

CFD calculations

In the CFD simulations discussed below, only a short section of the hot pipe is modelled. The inlet and the outlet boundaries are located one meter before and after the T-joint. At the inlet of the hot pipe, the mean flow velocity is 29 cm/s, which corresponds to a Reynolds number of $Re_D = 5.3\times10^5$. Therefore, the flow in the hot pipe is strongly turbulent. In the calculations presented below, a fully developed turbulent pipe flow is assumed as the boundary condition at the inlet. The temperature of the hot water at the inlet is 250 °C.

At the inlet of the cold pipe, the mean flow velocity is 1 mm/s, i.e., a very small amount of cold water is flowing into the pipe. The Reynolds number for the flow is $Re_D = 89$, which means that the flow in the cold pipe is laminar. Thus, a fully developed laminar pipe flow is used as the boundary condition at the inlet. The temperature of the cold water at the inlet is 50 °C.

Both the hot and the cold pipe are covered with a 60 mm thick layer of mineral wool. The temperature of the outer surface of the insulator is assumed to be 50 °C.

The flow equations and the conduction through the pipe wall and the insulator are solved with the Fluent 5.3 CFD code (Fluent 5 User's Guide 1998). Turbulence is modelled with the standard k-ε model and the Two-Layer Zonal (TLZ) model of Fluent. In the TLZ model, the one-equation model of Wolfstein is used for the turbulence in the near-wall regions, where the turbulent Reynolds number is small. With the aid of the TLZ model it is—at least in principle—possible to calculate the flow both in the high and in the low Reynolds number region. In the simulation, a quasi-stationary solution was found with a time-dependent calculation.

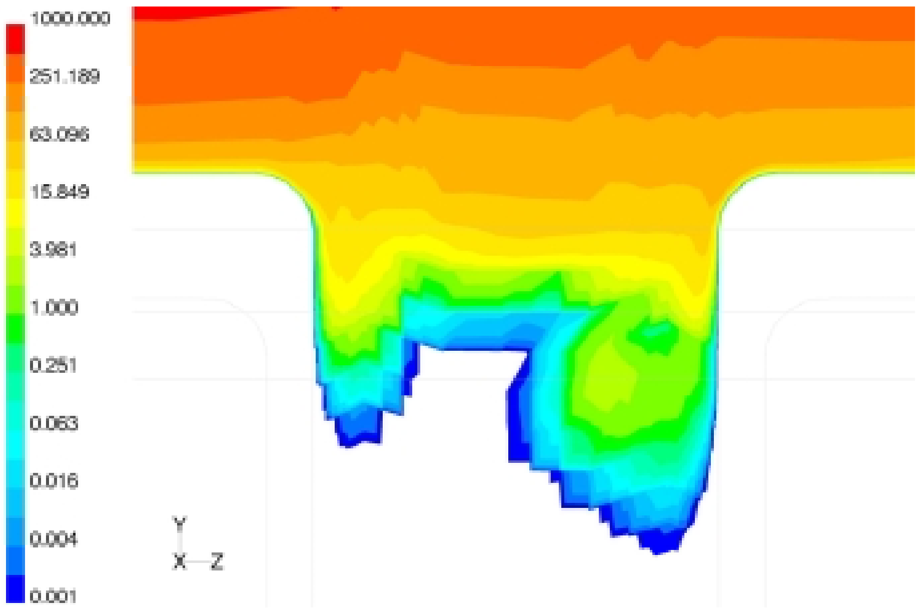

Figure 36. Turbulent viscosity ratio in the T-joint of the hot (horizontal) and cold (vertical) pipes. The cross-section shown is at the centre of the hot and the cold pipes. The direction of the gravitation is in the negative y-direction. Three horizontal cross-sections of the cold pipe are indicated. Results will be presented in the topmost horizontal cross-section.

The amount of turbulence transported from the hot pipe into the cold pipe affects the thermal and the momentum diffusion in the cold pipe near the T-joint. Therefore, an accurate prediction of the turbulence in this region is important. A measure for the level of the turbulence is the turbulent viscosity ratio $R_\mu = \mu_t / \mu_{lam}$, where μ_t is the turbulent viscosity and μ_{lam} is the laminar viscosity.

In figure 36, the turbulent viscosity ratio is shown in a vertical cross-section of the T-joint. A small amount of turbulence is transported from the hot pipe into the cold pipe. The turbulent viscosity in the cold pipe is, however, significant only in a short region near the T-joint. In the cold pipe, the effective viscosity decreases to the laminar level within a distance that is approximately equal to the diameter of the cold pipe.

In figure 37, the temperature and the flow velocity are shown in a vertical cross-section of the hot and cold pipes. In the cold pipe, a steep temperature gradient is formed within a region shorter than the inner diameter of the pipe. In the hot pipe, the hot water flowing at a high velocity of about 12 cm/s hits the rounded edge of the T-joint on the downstream side, where part of the hot water flows downwards into the cold pipe. On the downstream side of the T-joint, the cold water flowing upwards in the cold pipe meets the hot water flowing downwards. On the up-stream side of the T-joint, the cold water flows all the way up to the hot pipe, where it is mixed with the hot water. This small amount of cold water flowing into the hot pipe decreases the temperature on the downstream side of the T-joint by about 10–20 °C.

In figure 38, the temperature and the flow velocity are shown in a horizontal cross-section of the cold pipe near the T-joint (cf. figure 36). Near the wall on the downstream side, hot water with temperature of about 230 °C flows into the cold pipe. Near the wall on the upstream side, cold water with temperature of about 140 °C flows upwards.

In figure 39, the temperature of the inner surface of the pipe wall is shown. In the hot pipe, the wall on the downstream side of the T-joint is cooled down by the water flowing from the cold pipe. In the cold pipe, the temperature decreases within short distance from 250 °C to 50 °C. The width of the region with the temperature gradient is roughly equal to the diameter of the cold pipe. Recall that the temperature drop in the water occurred in a distance roughly equal to the radius of the cold pipe (see figure 37). The difference in the temperatures of the water and the pipe wall is caused by the good heat conductivity of the wall and the cooling effect of the water flowing into the cold pipe.

The results of the CFD calculations have been discussed in more detail by Pättikangas (2000).

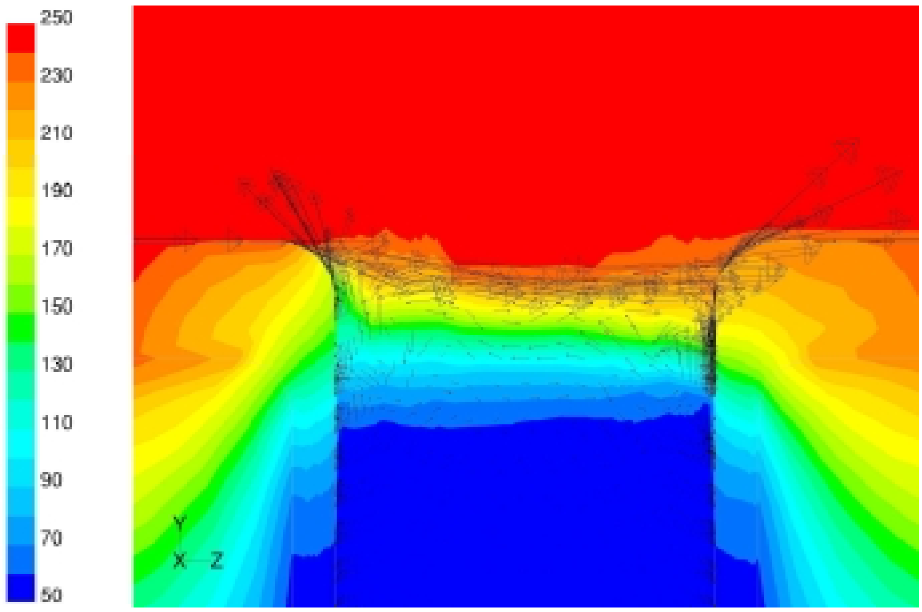

Figure 37. Temperature (contours, ℃) and flow velocity (vectors) in the T-joint of the hot (horizontal) and cold (vertical) pipes. The vertical cross-section shown is at the centre of the hot and the cold pipes. The gravitation is in the negative y-direction.

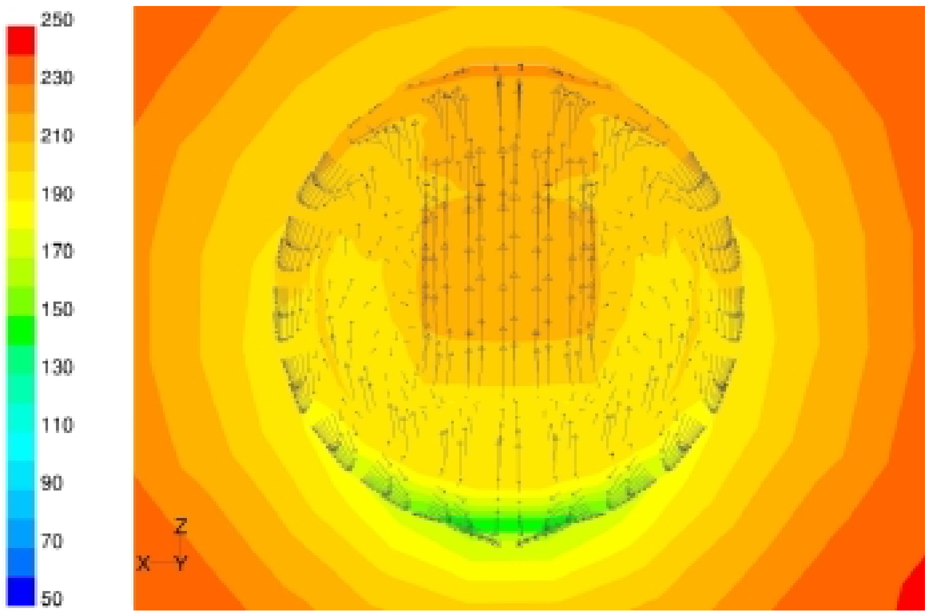

Figure 38. Temperature (contours, ℃) and flow velocity (vectors) in the cold pipe. The horizontal cross-section shown is at $y = -R_{hot} - 7$ mm, where R_{hot} is the inner radius of the hot pipe. The centre of the hot pipe is at $y = 0$.

Figure 39. Temperature (°C) of the inner surface of the pipe wall.

Structural analyses

Structural analyses were carried out using the commercial finite element (FE) code ABAQUS/Standard version 5.8 (ABAQUS Theory Manual 1998). The work for developing a transferring system from the data of CFD calculations to the input file of structural analyses is going on. Some capabilities are already in test use.

Preliminary calculations were carried out using the surface mesh of the CFD model and idealised boundary conditions. A part of the mesh can be seen in figure 40, where also the strain distribution is presented. There are about 8800 four nodded shell elements in the model. Temperatures at the inner surface of the pipe wall calculated by the CFD-code were transferred to the nodal points of the FE model. Temperature distribution through the pipe wall was assumed to be constant in these preliminary calculations. Room temperature was used as the stress free temperature.

Thermal strain distribution near the T-connection is shown in figure 40. This distribution clearly corresponds to the temperature distribution shown in figure 39.

The von Mises equivalent stress distribution due to thermal stratification is presented in figure 41. According to this calculation the stresses remain well below the yield strength of the steel. It should be noted that the details affecting the structural stiffness at the T-connection are not accurately modelled.

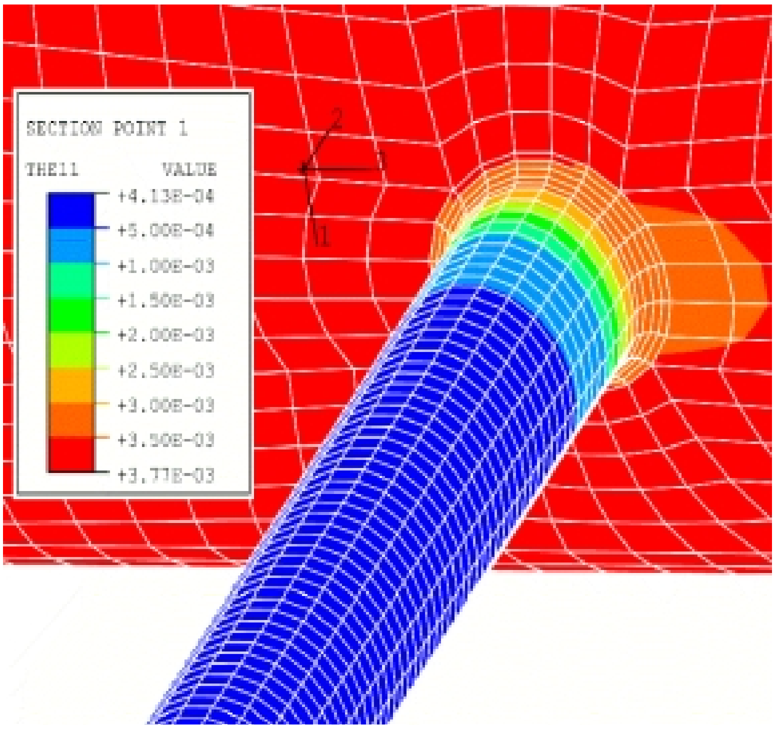

Figure 40. Thermal strain distribution (mm/mm).

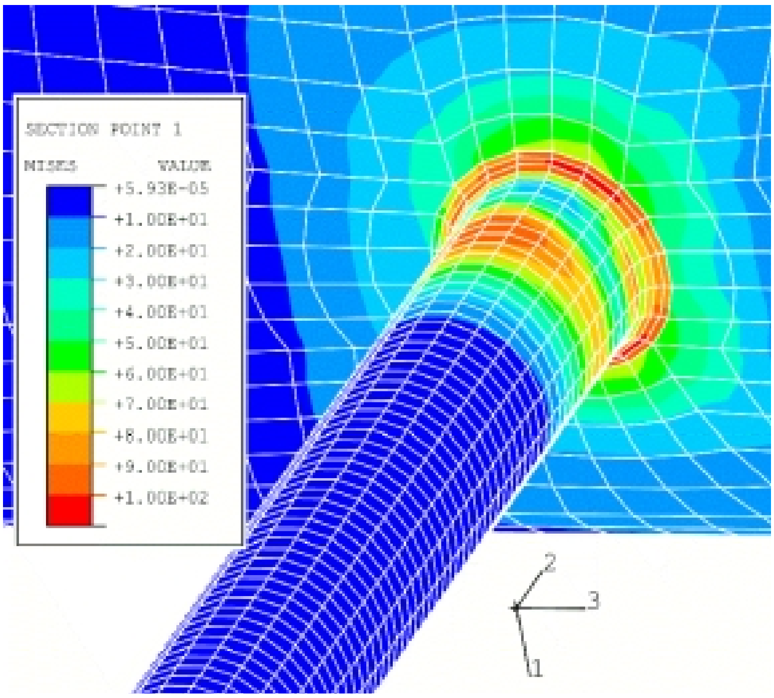

Figure 41. von Mises stress distribution (MPa).

In order to verify the numerical results obtained by using CFD and FE codes and the developed transferring tools, a stratification located clearly in the vertical pipe would be a convenient and practical case to study.

Summary and Conclusions

Investigations of thermal stratification in a T-joint of hot and cold pipes are in progress. An experimental set-up has been constructed based on the geometry of the connection line between hot and cold legs in the Loviisa NPP. Preliminary CFD calculations of the thermal stratification in the T-joint have been performed. The temperatures obtained from the CFD calculation have been used in preliminary structural analysis.

The construction of the experimental set-up for the thermal stratification tests is complete. The system consists of a large diameter horizontal and a small diameter vertical pipe that form a T-joint. The nearby PACTEL facility is used to produce heat and flow.

In the first phase of the experiments, thermal stratification in the vertical pipe with different boundary conditions for flow will be studied and the most interesting location from structural analysis point of view will be determined. In the second phase, a new test section with very extensive instrumentation concentrated on the interesting area, will be utilised. Later tests can focus, for example, on stratification in a horizontal pipe.

In the CFD simulation, the flow velocity and the heat transfer in a T-joint of hot and cold pipes were calculated with the Fluent code. In the hot pipe, the flow velocity was high and the flow was fully turbulent. A very small amount of cold water was flowing into the cold pipe, and the flow in the cold pipe was laminar. The Two-Layer Ional (TLZ) model of Fluent was used for modelling the turbulence. The heat transfer through the pipe wall and the surrounding insulator was also solved.

The CFD simulation shows that the temperature gradient is formed in the T-joint in a narrow region very close to the hot pipe. This result strongly depends on the boundary conditions that are assumed for the cold pipe. If one would assume a small outflow of cold water instead of the small inflow, the temperature gradient would be located further away from the T-joint.

Preliminary structural analysis was carried out using the surface mesh of the CFD model. Temperatures calculated by Fluent code were transferred to the input file needed for ABAQUS code.

The mesh used in CFD calculations is not necessarily optimal for structural analyses purposes. The development of the tools transferring the data of the CFD analysis to the structural analysis is under way. In order to verify the numerical results obtained by using CFD and FE codes and the developed transferring tools, a stratification located clearly in the vertical pipe would be a convenient and practical case to study.

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5.3 STIN special report 2

Constraint correction and transferability of fracture mechanical parameter

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Abstract

The use of fracture mechanics in design and failure assessment is to some extent impeded by the difficulties of quantifying the structure related constraint. It is well known that specimen size, crack depth and loading conditions may effect the materials fracture toughness. In order to safeguard against these geometry effects, fracture toughness testing standards prescribe the use of highly constrained deep cracked bend specimens having a sufficient size to guarantee conservative fracture toughness values. An example of one of the more advanced testing standards providing a method to determine such a "base line" fracture toughness characterization for brittle fracture is the so called Master Curve standard ASTM E1921-97 [1]. These "base line" toughness values have one weakness. When applied to a structure with low constraint geometry, the standard fracture toughness estimates may lead to strongly over-conservative estimates. In some cases this may lead to unnecessary repairs or even to a too early "retirement" of the structure. In this work, a connection between the constraint parameter called T-stress and the Master Curve transition temperature T_0 is developed. As a result, a new tool to assess low constraint geometries with respect to brittle fracture is obtained.

Introduction

In the case of brittle fracture, essentially three different methods to quantify constraint have been proposed, J-small scale yielding correction (SSYC) [2, 3], Q-parameter [4, 5] and the T_{stress} [6–9]. A comprehensive summary of the three methods has been given e.g. by Nevalainen [10] and will not be repeated here. Although they are more sophisticated, the SSYC and Q-methods require the use of very detailed elastic-plastic finite element analysis, whereas the simpler T_{stress} only requires an elastic analysis. The SSYC method tries to directly quantify the effect of constraint on toughness by scaling the stress distribution in front of the crack to correspond to a high constraint situation. The successful use of this method requires the stress contours to be self-similar. Often, for low constraint geometries this is not the case. The SSYC correction has recently been further developed by connecting it to the local approach of fracture and the Master Curve [11], but its use requires a calibration with a low constraint geometry. The Q-parameter and the T_{stress} both give a quantitative value for the constraint, but the connection to toughness is lacking.

Here, a connection between the T_{stress} and the Master Curve transition temperature T_0 is attempted. The connection is based on the assumptions presented in figure 42. It is assumed that, if the constraint can be described by the T_{stress}, then the shape of the fracture toughness temperature dependence is not significantly affected and the constraint differences can be effectively described as a shift in the transition temperature T_0. If the constraint is described by the Q-parameter (large scale yielding), then also the shape of the fracture toughness temperature dependence is affected and a more detailed analysis is required. However, even then, quantifying the constraint in term of T_{stress} will be conservative. This means that if the assumptions made here can be verified and the T_{stress} can be connected with the Master Curve transition temperature T_0, a new tool to assess low constraint geometries with respect to brittle *fracture is obtained.*

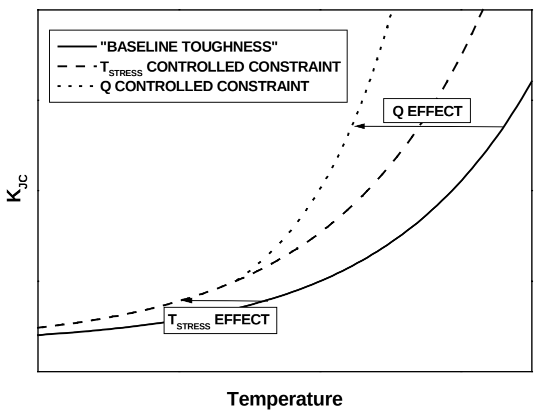

Figure 42. Schematic representation of assumed effect of T_{stress} and Q.

It is imperative that the verification of the assumptions and quantification of the constraint is performed with a well-characterized test geometry. One such geometry is the shallow cracked bend specimen, where both, the K_J analysis, as well as the T_{stress}, are well known. The bend geometry remains also in contained yielding to comparatively high K_J-values, thus indicating a T_{stress} controlled constraint description. For this specimen geometry there is also a considerable amount of data available in the literature, omitting the need for any further experimental work. It is the intent of this work to quantify this increase in toughness in terms of a shift in the Master Curve T_0 transition temperature.

Materials

The data for the analysis was collected from different literature sources, focussing on fracture toughness results from three-point bend specimens with varying a/W values.

One pre-requisite for the data was that it should be possible to analyze with the Master Curve method. The list of materials and specimen dimensions is given in table 4.

One problem connected to shallow flaw fracture toughness results, is the lack of a common test standard for these geometries. This has led to the use of various different estimation methodologies, being based on either area under load – load-line-displacement curve (A_{PLLD}), area under load – crack-mouth-opening curve (A_{PCMOD}) or crack-tip-opening-displacement (CTOD). Besides differences in basic measurements, also the J-integral (K_J) estimation equations may show considerable variability, which in some cases necessitated a slight re-evaluation of the basic data.

Table 4. Materials.

Source	Material	σ_Y MPa	a/W	B mm	b mm
Sumpter [12]	Mild Steel	235	0.05 - 0.7	23 & 25	15 - 28
Link & Joyce [13]	A533B Cl.1	580	0.1 - 0.55	25 & 89	23 - 74
Sorem & al. [14]	A36 Steel	248	0.15 & 0.5	13 & 32	6 - 32
Theiss & al. [15]	A533B Cl.1	400	0.1 & 0.5	51 - 153	48 - 90
Theiss & al. [15]	A533B Cl.1	432	0.1 & 0.5	101	60 - 91
Smith & Rolfe [16]	- " -	- " -	0.1 - 0.5	20 - 32	8 - 22
Dawes [17]	BS4360 G50C	360	0.2 & 0.5	25	13 - 25
Timofeev & al. [18]	15X2MFA	530	0.1 - 0.5	50 & 100	25 - 50
Dlouhý & al. [19]	SAF 2205	548	0.22 - 0.5	25	23 - 25
Sumpter [20]	HY80 W [TS]	650	0.1 & 0.3	35	25 - 31
Sumpter [20]	HY80 W [TL]	650	0.1 & 0.3	35	49 - 63
NESC- SC1 [21]	A508 Cl.3	570	0.1 & 0.5	25	13 - 22
Nevalainen & al. [22]	A533B Cl.1 D	580[1]	0.03 - 0.49	10	5.1 - 9.7
Kirk & al. [23]	A517 G70	300	0.13 - 0.53	10 - 51	5 - 45
Ruggieri [24]	H.S. Steel	663	0.1 - 0.5	15	7 - 13
Sumpter [25]	HY80 W D	760[2]	0.2 & 0.3	50	35 & 40
Sumpter [25]	HY80 W D	800[3]	0.15 - 0.32	50	34 - 43
Roos & Eisele [26]	10MnMoNi 55	630	0.1 & 0.55	25	22 - 45

[1] Estimated dynamic yield strength (σ_Y = 480 MPa).
[2] Estimated dynamic yield strength (σ_Y = 660 MPa).
[3] Estimated dynamic yield strength (σ_Y = 705 MPa).

Analysis

Master Curve

The Master Curve analysis followed in principle the ASTM E1921-97 standard [1], but the nature of some of the data sets required a more flexible analysis method. Two levels of censoring were applied. First, all data referring to “non-cleavage” (ductile end of test) were prescribed δ_i = 0. Second, all data violating the specimen size validity criterion

(Eqn. 1) were assigned the toughness value corresponding to the validity criterion and given $\delta_i = 0$.

The specimen size validity criterion (taken similar as in the ASTM E1921-97 standard) has the form

$$K_{JC} \leq \sqrt{\frac{b \cdot \sigma_y \cdot E}{M \cdot \left(1 - \upsilon^2\right)}} \tag{10}$$

In Eqn. 10, σ_y is yield strength, E is the modulus of elasticity, the controlling dimension is the ligament size b and the size criterion constant M = 30. The use of the ligament, b, and not the crack depth, a, as the controlling dimension also for the shallow cracks, was based on the assumption, that any possible front face effects of the shallow flaw specimens is essentially accounted for and described by the T_{stress}. The limit of contained yielding, which is the parameter that is assumed to really determine the measuring capacity of the specimen, is controlled by the ligament. As a check of the above assumption, the shallow crack specimens were also evaluated using crack length a as the controlling dimension. Since this is equivalent to increasing the size criterion M, with respect to b, also the deep flaw specimens were analysed using M = 120.

T_{stress}

The T_{stress} is normally expressed in terms of the biaxiality ratio β, which connects the T_{stress} with the linear elastic stress intensity factor K_I (Eqn. 11). The biaxiality ratio is purely geometry related, so the T_{stress} will be a function of stress level.

$$\beta \equiv \frac{T \cdot \sqrt{\pi \cdot a}}{K_I} \tag{11}$$

Similarly to having several different η factor solutions, T_{stress} solutions for the three point bend specimen show some variability. For consistency, all the data were analyzed using only one T_{stress} solution. The solution by Sham [27] was selected as reference.

A fifth degree polynomial was fitted to the Sham T_{stress} solution, which is only given for specific a/W ratios. The problem with this solution is that the T_{stress} is directly dependent on the nominal stress level of the specimen. Since this was normally not known for the data sets, a modified procedure suggested by Sumpter [28] was used. This consisted of presenting the T_{stress} related to the materials yield strength (Eqn. 12).

$$\frac{T}{\sigma_Y} = \frac{\beta \cdot Y \cdot \sigma}{\sigma_Y} \tag{12}$$

In Eqn. 12, Y is the stress intensity factor geometry function. Making the quite realistic assumption that the test specimens usually failed close to plastic limit load, enables one to replace the nominal stress σ with the plastic limit stress σ_L, which was taken as [28]

$$\sigma_L \approx 2.184 \cdot \sigma_Y \cdot (1 - a/W)^2 \tag{13}$$

This gives an approximate T_{stress} solution related directly to the materials yield strength (Eq. 14).

$$\frac{T}{\sigma_Y} \approx \beta \cdot Y \cdot 2.184 \cdot (1 - a/W)^2 \tag{14}$$

The resulting fifth order polynomial has the form

$$\frac{T_{stress}}{\sigma_Y} \approx -1.13 + 5.96 \cdot \frac{a}{W} - 12.68 \cdot \left(\frac{a}{W}\right)^2 + 18.31 \cdot \left(\frac{a}{W}\right)^3 - 15.7 \cdot \left(\frac{a}{W}\right)^4 + 5.6 \cdot \left(\frac{a}{W}\right)^5 \quad for \quad \left[\frac{a}{W} \le 0.9\right] \tag{15}$$

Results and discussion

As an example, the Master Curve analyses of one material is presented in figure 43. The different T_0 estimates are compared in figure 44. Practically all the data fall within a ± 10ºC scatter band and no specific trend, for the more severe censoring criteria to produce higher T_0 values, are seen. Thus, the assumption that the specimens measuring capacity is controlled by the ligament size also for shallow cracked specimens and M = 30 is an adequate censoring limit, is supported by the data.

Besides validating the censoring limit, figure 44 also support the assumption that the shape of the Master Curve remains essentially unaffected by T_{stress}. If the shape would change, the more severe censoring criteria should produce systematically different T_0 values. Since this did not occur, the shape must have remained the same. In order to study whether any similarities exist between the T_0 behavior of the different data sets, the results were also expressed in terms of the temperature difference between the shallow crack T_0 and the deep crack T_{0deep}. The result is plotted as a function of relative crack depth in figure 45. The data clearly shows the expected trend of increasing toughness with decreasing crack depth, but the figure reveals more. The data was

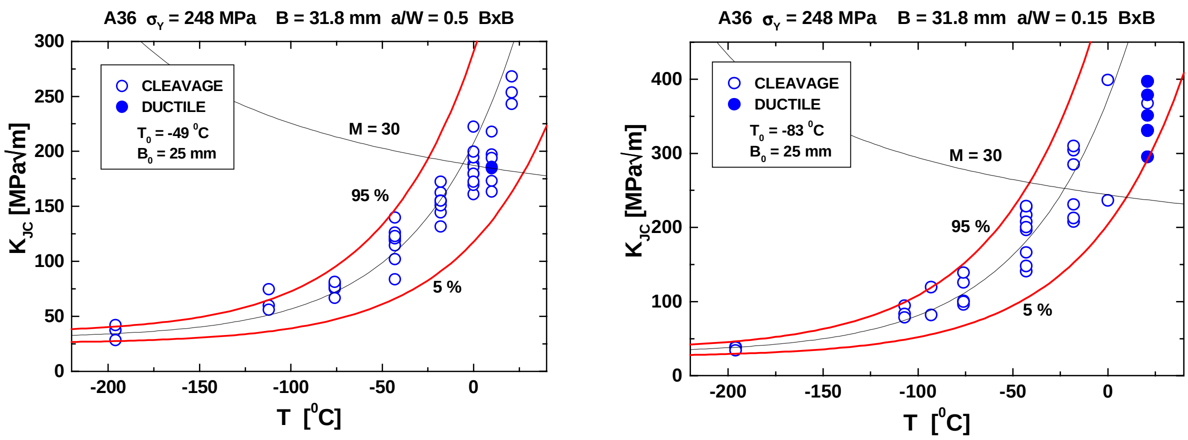

Figure 43. Master Curve analysis of A36 Steel data by Sorem & al. [14].

divided into three groups depending on their yield strength at the T_0 temperature. The data shows that the higher the yield strength of the material, the larger the shallow crack effect is, i.e. the lower is the constraint. The picture changes, when the data is plotted against the estimated T_{stress} of the data sets (figure 46). The data still shows the expected trend of increasing toughness with increasing negative T_{stress}, but the effect of yield strength is effectively removed. Figure 46 indicates that positive T_{stress} has an insignificant effect on fracture toughness, whereas for negative values of T_{stress}, the Master Curve T_0 changes nearly linearly with T_{stress}. Since the effect of large scale yielding related loss of constraint (described with Q), probably also affects the results in figure 46 to some degree, it is better not to define a mean T_{stress} dependence for T_0, but to make it a little conservative. This results in a simple linear relation giving the effect of T_{stress} on T_0 (Eqn. 16).

$$T_0 \approx T_{0deep} + \frac{T_{stress}}{10\,MPa/°C} \quad : for\ T_{stress} < 0 \qquad (16)$$

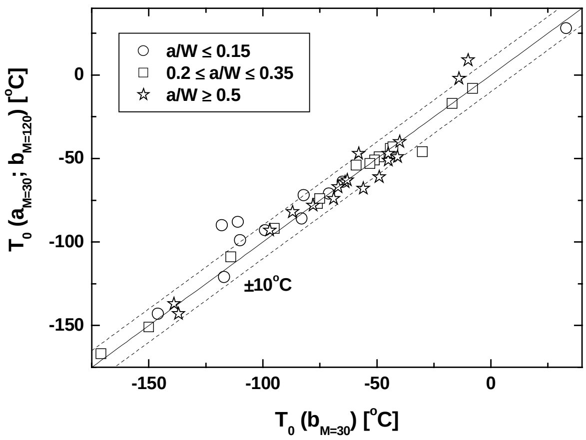

Figure 44. Comparison of T_0 estimates based on different censoring criteria.

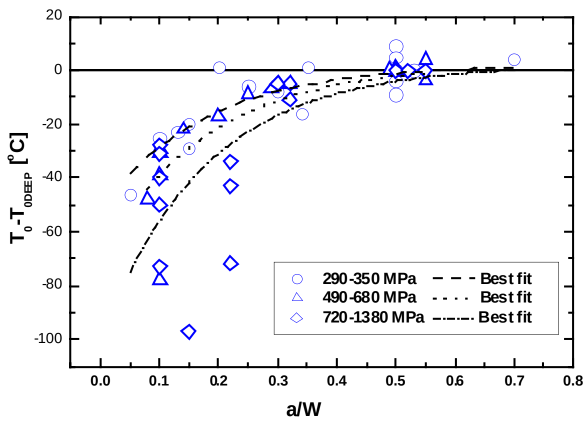

Figure 45. Effect of relative crack depth, on the T_0 transition temperature.

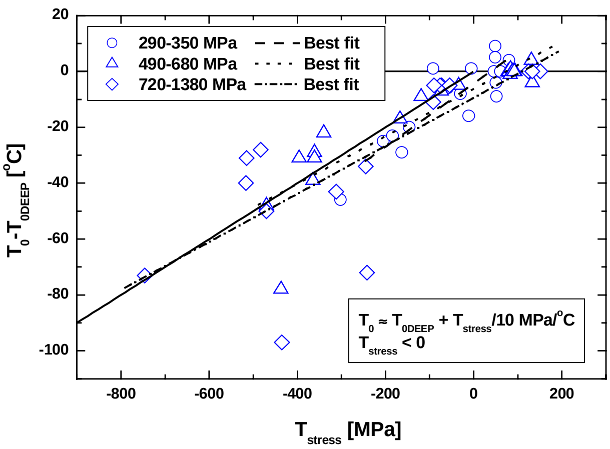

Figure 46. Effect of T_{stress} on the T_0 transition temperature.

The present result can also be transformed into a simple approximate constraint correction directly for K_{JC}. Combining Eqn. 16 with the Master Curve temperature dependence [1], a constraint correction for a single value, fracture toughness, can be approximated as (Eqn. 17).

$$K_{JC} \approx 20\,MPa\sqrt{m} + \left(K_{JCdeep} - 20\,MPa\sqrt{m}\right)\cdot exp\left(0.019\cdot\left[\frac{-T_{stress}}{10\,MPa}\right]\right) \qquad (17)$$

$$for\ T_{stress} < 0$$

As a check of the validity of Eqn. 17, a "blind" test was performed on a data set not included in the original analysis. For this purpose a data set generated by Betegón was used [29]. For the application of Eqn. 17, T_{stress} was estimated with Eqn. 15. The comparison is presented in figure 47.

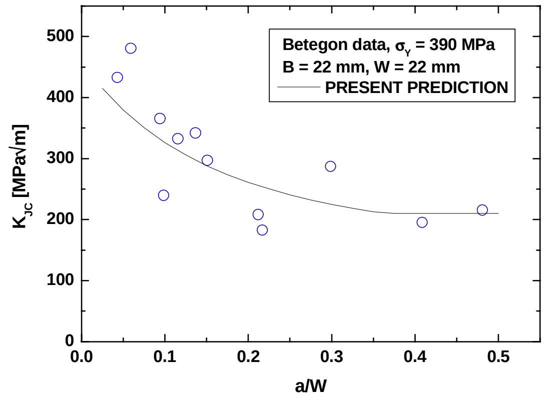

Figure 47. Comparison of the constraint correction prediction given by Eqn. 13 with Betegón data[29].

Considering all the simplifications and approximations that had to be performed in the analysis of the data sets, the overall results are surprisingly straightforward. The general trends and conclusions summarized by Eqs. 16 and 17 are very well corroborated by the data.

An important fact to note is that Eqs. 16 and 17 should only be used to correct high constraint fracture toughness results to lower constraint geometries, never to correct low constraint fracture toughness results to higher constraint geometries.

Even though only bend geometries were included in the present study, the results should be equally applicable for tension geometries. These loose contained yielding earlier than bend specimens and the effect will just be to make Eqs. 16 & 17 more conservative for these geometries. A more precise description of tension geometries and bend specimens in large scale yielding requires the use of the Q-parameter or some similar method capable of describing large scale yielding effects. This appears to be a natural next step in the development of the Master Curve technology.

Summary and Conclusions

In this work several data sets with varying crack depth, three point bend specimen, brittle fracture toughness results have been analyzed with the Master Curve method and interpreted in terms of the individual T_{stress} values of the data sets. As a result, a simple tool for the application of the Master Curve technology to low constraint geometries is obtained. The fracture toughness of the structure will normally be conservatively estimated by the developed equations, so the integrity of the structure is not in jeopardy even when the structure-specific constraint is accounted for.

Based on the study, the following conclusions can be made:

- Crack depth does not have a significant effect on the temperature dependence, nor scatter of brittle fracture toughness.

- The specimens' measuring capacity is controlled by the ligament size also for shallow cracked specimens.

- Positive T_{stress} has an insignificant effect on fracture toughness, whereas for negative values of T_{stress}, the Master Curve T_0 changes nearly linearly with T_{stress}.

- The determined relation between T_{stress} and T_0 provides a simple tool for the application of the Master Curve technology also to low constraint geometries.

- The methodology can be further improved by including plasticity effects into the analysis.

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5.4 INSMO special report

Production of artificial defects and their applicability to qualification of ultrasonic inspection

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Abstract

The qualification of non-destructive test systems shall demonstrate in a realistic and reliable way the performance of different parts of the application. An inspection trial known as "practical test", where the component and accessibility as well as the defects resemble those of the real inspection situation, can be used as a very effective and concrete evidence of the capability of the system. Therefore the qualification activity but also the development of NDT applications require nowadays very often production of test samples and also manufacturing of different defects or defect simulations in them.

Ultrasonic testing is usually used to detect different types of cracks. Real cracks are difficult to produce in a controlled way and therefore artificial defects with simple geometry are often used as reflectors to simulate the defects in the component. The ultrasonic response of the reflectors should be similar to real defects as much as possible. In practice the simulation is not always ideal and thus the imperfection and limitations should be known. The work described in this paper covers searching and testing of suitable defect and reflector manufacturing techniques for qualification sample production and examining their suitability and applicability in different situations.

Introduction

The performance of non-destructive testing (NDT) methods was studied and measured using extensive round robin projects during late 1980's continuing until beginning of 1990's. The research was mainly financed by EU and is known as PISC (Programme for the Inspection of Steel Components) programme. During the programme inspection of different nuclear power plant (NPP) components was considered and in the most cases ultrasonic inspection was the main method to be used. PISC programme showed that the performance of inspections was not always as good as believed and some actions should be taken to enhance the reliability. The qualification of inspection methods was seen internationally as an effective way to enhance the reliability. The co-operation of

European countries to create a NDT qualification system is known as ENIQ (European Network for Inspection Qualification). Also in US the qualification issue has been important during 1990's and requirements to proof the inspection performance have been included in the ASME Code.

Qualification of NDT methods is used to demonstrate the performance of system, procedure and personnel. ENIQ has designed the European methodology, that consists of two parts: practical assessment and technical justification (ENIQ 1997). Practical assessment is conducted on simplified or representative test pieces resembling the component to be inspected. The defects included in the test piece may either be known to the inspection team (open test) or unknown (blind test). The technical justification is a written document where all the evidences showing the capability of the inspection are presented. According to the ENIQ methodology the qualification should always include technical justification and practical assessment is completing the case. The requirements set for the practical assessment depend on one hand on the demands of the inspection itself and on the other hand on the level of the evidences presented in the technical justification.

Practical test sets a need for test blocks, that shall be representative when considering the component design, materials, production techniques etc. Thus the production is in many cases expensive and time consuming. As examples two test blocks used in real qualification situations are shown in the figure 48. The first is produced for qualification of primary circuit pipe-to-nozzle welds and the second for qualification of primary circuit pipe-to-main gate valve nozzle weld.

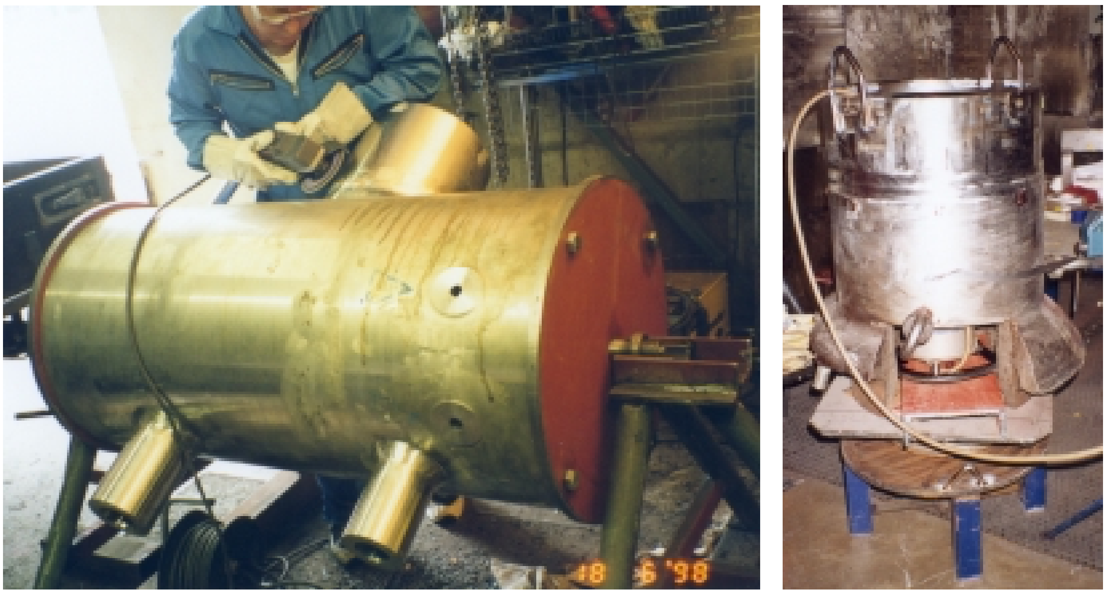

Figure 48. Examples of qualification blocks used for ultrasonic inspection qualification of NPP primary circuit welds.

If the test block is manufactured using same materials and production technique as the real component in the plant it can verify in very comprehensive terms the capability of inspection equipment and procedure. But also the production is often very complicated because the materials used in a running NPP are difficult to purchase and the know-how about the welding procedures and other production techniques may be hard to track. In addition the costs of the test block are increased further by the need to include the necessary defects in the test block either during the manufacturing process or after it.

The defects should be as representative as possible when the properties of the known or postulated defects of the component are considered. Only in very few cases abandoned components with real defects are available and can be used as test blocks. Usually the defects have to be manufactured intentionally. Their type, geometry, location and size should be controlled and known by the organiser of the qualification test. If the type of an artificial defect cannot be the exactly similar with the real defect, which is often the case in the real life, the behaviour of the defect seen by the inspection method should be as realistic as possible.

Production techniques of test defects

When defects are produced into qualification test samples several requirements are usually set for them. These requirements are necessary to ensure that the qualification is testing the most important features of the inspection and is producing evidences showing the required capability of the inspection system. In the most cases it is important that the size, location and orientation of the defects to be produced can be controlled according to the characteristics of the existing or postulated defects. Usually some variation of these parameters is needed to be able to measure the capability and limits of the inspection system. Often it would be important to control and imitate the shape and surface roughness of the cracks that can have large variation as shown in figure 49. These crack shape and surface conditions have often considerable effects on ultrasonic reflectivity and should therefore be carefully considered.

One requirement set for the test defect production is that it shall not hinder the inspection or create disturbances in the material surrounding the defect. If the production of the defect affects the material round itself these disturbances can reveal the existence of the defect to the inspector even when he cannot indicate the defect itself. On the other hand it is possible that such disturbances mask the defect in such a way that proper judgement about the existence and size of the defect are impossible.

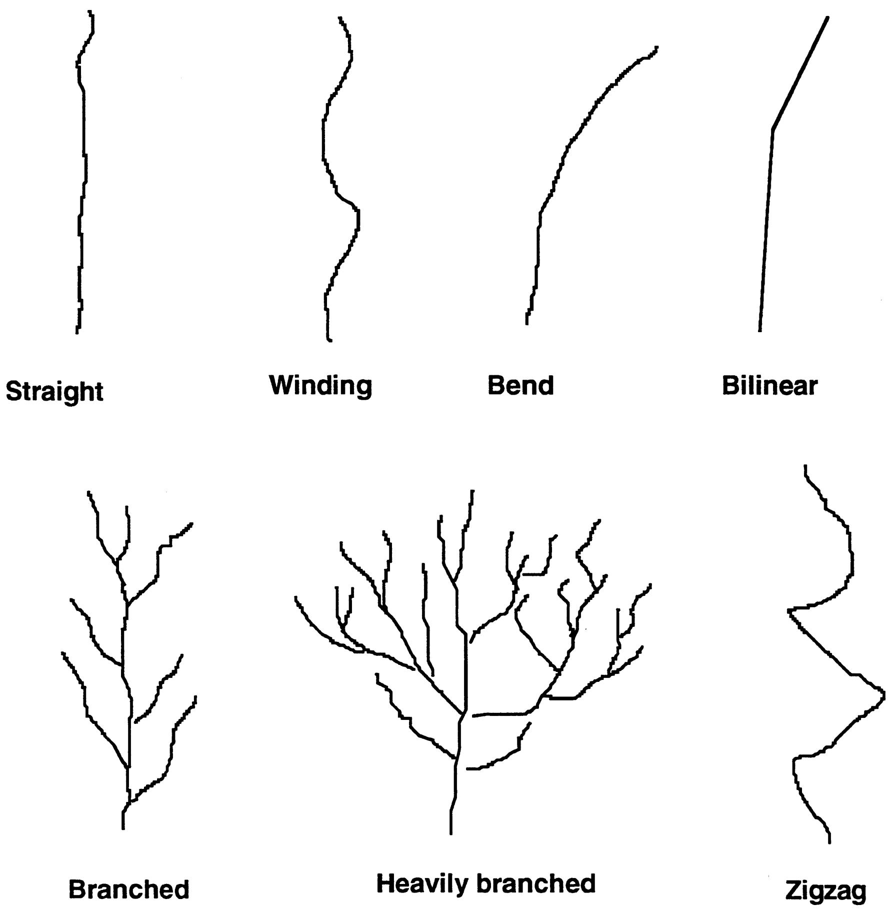

Figure 49. Schematic illustration of different types of crack shapes (Ekström & Wåle 1995).

When the defects used in the ultrasonic qualification blocks are considered two approaches are used to produce them. Either there are realistic defects produced or implanted into the sample or much simpler notches are used as reflectors that simulate the defects.

The realistic defects are of course much better simulations for the real cracks expected to be in the component than simple notches. But the production of cracks is often quite demanding task that requires a lot of know-how and experience. Also the parameters of the defect like size and orientation may be hard to control exactly during the manufacturing process. In addition it may be necessary to perform destructive testing to ensure the final values of these parameters. This is normally possible only afterwards when the validation actions themselves are performed. Typically cracks are produced by mechanical or thermal fatigue processes, creating cracks using special welding parameters or applying chemical processes. One possibility, that is quite commonly used, is to implant small sample that includes a crack into test block using skilful welding technique.

In many cases notches machined by mechanical means or by electro-discharge machining (EDM) can be used quite satisfactorily as ultrasonic reflectors that simulate simple defects. The location, size, geometry and orientation can be exactly controlled and the manufacturing is rather cheap and straightforward without too many risks that could jeopardise the whole block. Notches have typically straight wall surfaces that reflect ultrasound like mirrors which can often simulate fatigue crack quite well. On the other hand in the case of stress corrosion cracks the defects are often winding, zigzagged or branched and the simulation of the reflectivity properties by notches is much poorer.

Defects produced in the current project

During the current project several series of artificial defects are produced. The objective has been to produce typical defects that could be used as ultrasonic reflectors in the qualification blocks. One dimension of the work has been test and experiment different manufacturing processes and features to find out how well these are controlled and where are the limits of the methods used. The other dimension has been to measure the ultrasonic responses of these reflectors and try to compare and assess artificial reflectors in comparison to real defects. Thus the final objective is to form conclusions about the applicability of the reflectors in different qualification applications.

Several types of reflectors have been manufactured using EDM. This method is seen potential because reflectors to be produced can be controlled precisely and the shape of the notch can be formed in some limits by the design of the working electrode. One set of defects is produced by special welding technique that created hot cracks. Also a couple of fatigue cracks are manufactured in the vicinity of the weld root.

All the defects are manufactured in austenitic stainless steel material. This material is commonly used in primary circuit components of NPPs. The ultrasonic signals received in this material have typically much lower signal-to-noise levels compared to ferritic materials (Crutzen et al. 1996) and experiments using such material are thus seen crucial when NPP applications are considered.

All the defects and reflectors available so far for ultrasonic measurements are shown in the table 5. The 13 hot cracks are produced by Fortum and handed over to project for measurements. All the other defects are manufactured by the project. A typical test sample of the project is shown in the figure 50.

Table 5. The defects available for measurements in the project.

Defect type	Position	Manufacturing	Description	Number	Special Interest
EDM notch	Base mat.	Disc electrode	Conical 4° - 12°	6	Sharp tip
EDM notch	Base mat.	Plate electrode	Slit 0.6 - 0.7 mm	6	Easy manuf.
EDM notch	Base mat.	Tube electrode	Row of holes	3	Reflectivity
EDM notch	Base mat.	Plate electrode	Slit 0.2 mm	4	Maybe standard
EDM notch	Base mat.	Plate electrode	Slit 0.1 mm	2	As narrow as possible
EDM notch	Weld	Plate electrode	Slit 0.2 mm	4	Maybe standard
Fatigue crack	Weld	Mechanical	Crack at weld root	2	Realistic defect
Hot crack	Weld	Weld process	Crack in weld	13	Realistic defect

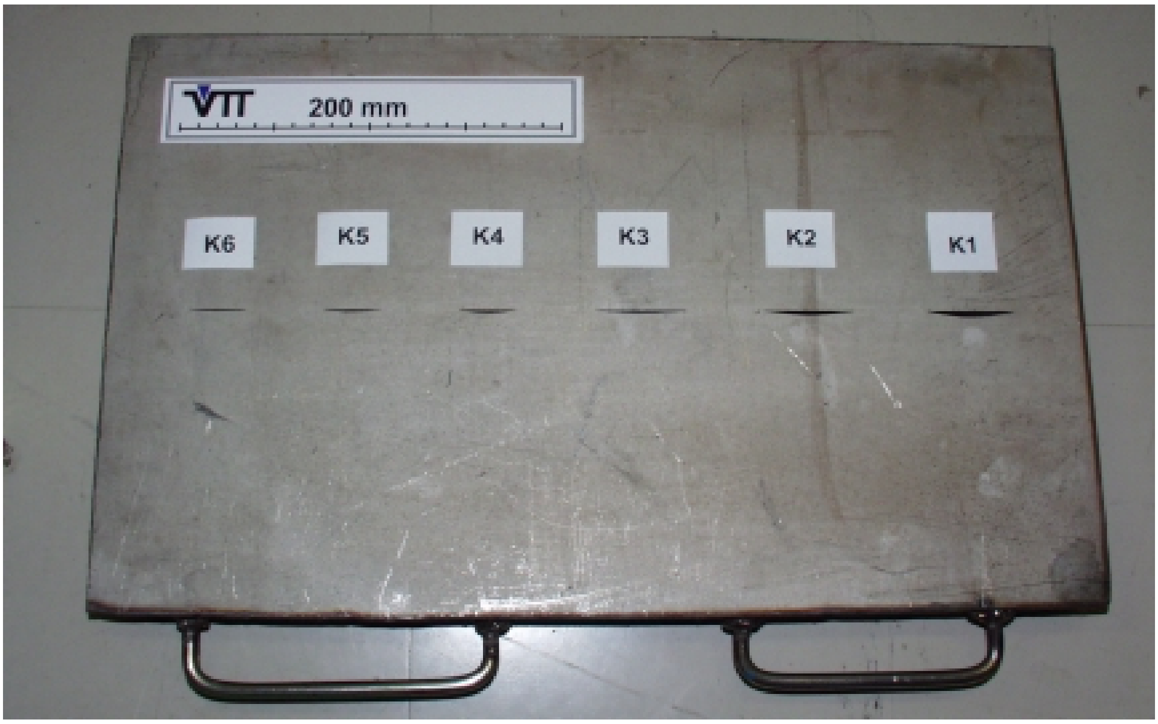

Figure 50. Test sample including ultrasonic reflectors produced by EDM method.

Ultrasonic inspection of the project test samples

Ultrasonic signals are measured and recorded using mechanised automatic inspection system. The data acquisition system is SUMIAD III and the probe is scanned by SAM equipment (Scanning Acoustic Microscope). Thus all the information available in the signals received can be recorded using very dense and accurate measurement grid. The probes applied are typical for real in-service inspections to have comparable results with field applications.

The measurement results are analysed using MASERA program that is able to produce amplitude level plots of received signals in different projections (ultrasonic C-, B-, and D-plots). In the example of the figure 51 are shown the indications recorded from the tree largest reflectors (K1–K3) of the test block presented above in the figure 50. From these plots the signal amplitudes and the co-ordinates of their origin can be measured to analyse the signal-to-noise ratios and to determine the length and height of each reflector.

At the moment of writing this report only the three first reflector series given in table 5 are measured and analysed. Thus the results presented here are concerning only those.

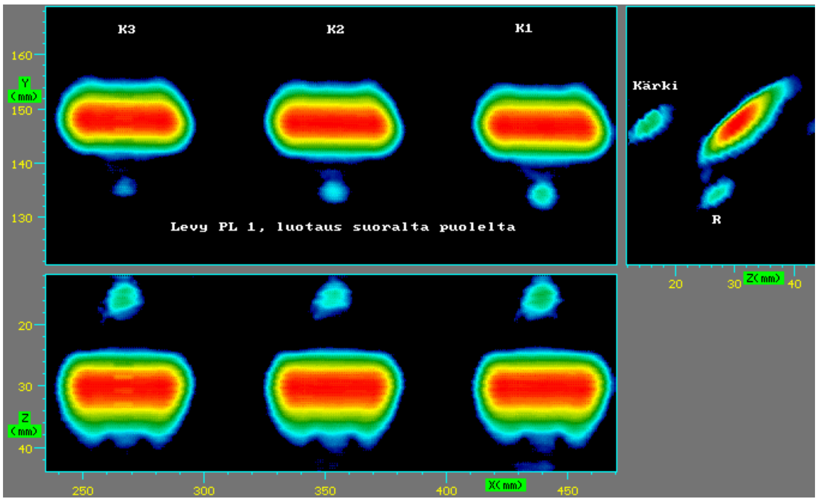

Figure 51. Amplitude level plots of three artificial defects. The recorded amplitude levels of received ultrasonic echoes are presented in different colours. Three projections are included thus the co-ordinates of the echo origins can be defined.

All the artificial defects could be detected easily using corner echo with high signal-to-noise ratio. The influences of the size, geometry and orientation of reflectors on the signal patterns (indications) and levels could be noticed and measured. The length measurements of the reflectors from the recorded data using normal field procedure gave very sufficient accuracy the maximum errors being in the order of 10 %. The crack tip diffraction signal could be recorded with sufficient signal-to-noise ratio nearly from all of the measured reflectors. Using this signal the location of the crack tip could be seen very exactly and the determination of the reflector height was straightforward.

The measurement results produced basic information about the applicability of the reflectors. Because the reflectors were manufactured into base material all the signal types arising could be carefully recorded and may be used as a clue when reflectors are applied in more complicated situations. Also the measured echo amplitudes and their variations because of size and geometric factors are valuable information when reflectors for real test blocks shall be designed.

Objectives of the second half of the project

Some more artificial defects will still be produced during the project. When designing these different reflector types, geometry, location and orientation will be considered. Also attention shall be paid on new manufacturing techniques. Component and construction details that can have influence on the functionality of the reflectors could also be important factor to include in the research program.

To assess the applicability of the artificial reflectors their ultrasonic response should be compared to signals received from real defects. Therefore one intention of the project is to acquire, if possible, a set of real defects and use these as reference material. Also recorded data from real inspection can be useful for comparison and assessment but the final and exact information about the defect details may remain imperfect when this approach is used.

Some destructive test will be necessary at the end of the project to examine and measure the exact geometry and dimensions of the reflectors and defects produced as well as those of the possible real defects available. Also metallurgical effects caused be defect manufacturing processes may be necessary to consider and some examinations about these can be needed. Final conclusion about the manufacturing processes can be drawn only after this phase.

The final stage of the project is to compile conclusions giving some level of guidance for selection and design of reflectors and artificial defects to be used in qualification blocks of ultrasonic inspection. It shall include information about the manufacturing and application possibilities of different reflector types. Also the possibilities and limits of the reflectors to imitate certain defect types and their features should be considered from the ultrasonic inspection point of view.

Conclusions

When test blocks are manufactured for qualification purposes appropriate test defects must be included to measure the performance of the NDT method. In the case of ultrasonic inspection suitable test blocks that include real defects are very hard to acquire and in practice difficult to use in qualification exercises. The production of test blocks using artificial defects is less expensive and time-consuming, larger number of defects including limit cases can be included and relevant material and geometrical selection can be made.

The current project is producing information about the applicability of artificial defects and reflectors. Different defect types and features of defects are considered. Many defect production techniques are used and experience about the possible applications as well as ultrasonic measurement results are reported. At the final stage of the project an assessment about the ultrasonic properties of the artificial defects and reflectors compared to those of real defects will be made.

The time span of the project is four years and it includes plenty of experimental work. So far the work completed has included mostly design and manufacturing of the reflectors. At the moment the main emphasis of the work is on the recording and analysing ultrasonic data. Destructive testing is seen necessary at the end of the project to examine and verify the details of defects and reflectors produced.

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5.5 KOTO special report

Effect of creep on fuel behaviour in RIA

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Abstract

Creep is a high temperature deformation mechanism that becomes significant when the temperature exceeds about half of the material absolute melting point. An earlier study at VTT shows that creep is a significant deformation mechanism in certain reactivity initiated accidents (RIA). So, to be able to realistically simulate RIA creep has to be included in the simulation.

VTT Energy is using the SCANAIR computer code to analyse the behaviour of nuclear fuel in accident conditions. The code is meant to simulate a reactivity-initiated accident in a light water reactor. SCANAIR was designed to handle the closely interconnected thermal-mechanical calculations that are needed to simulate the accident behaviour of a fuel rod. The code simulates the thermal-mechanical behaviour of a single nuclear fuel rod with numerical models. A finite element model is used in the mechanical module of SCANAIR to simulate the mechanical behaviour of the rod. The mechanical module is designed to describe the highly nonlinear mechanical behaviour of the fuel rod taking into account elasticity, plasticity, thermal expansion, cracking, and fission gas swelling. Also a creep model has been implemented in SCANAIR module at VTT Energy.

This paper briefly describes the implemented creep model in the SCANAIR code. A simple example calculation is also made with SCANAIR to show the effect of the creep on RIA calculation.

Introduction

VTT Energy is using a computer code called SCANAIR to simulate nuclear fuel behaviour in fast transients, such as reactivity initiated accidents (RIA). RIA is usually thought to be triggered by control rod ejection. A rapid rise in reactivity and potentially large power surge in the nuclear fuel are typical of RIA. The power pulse length in RIA is some tens of milliseconds depending on reactor design making RIA a very fast transient. The power surge causes a rise in temperature of the fuel. The pellets rapidly expand due to thermal expansion and swelling during the power pulse. The expansion of the pellet closes the gas gap between the pellet and the clad if a gap still exists. The closure of the gap leads to mechanical interaction between the pellet and the clad. This mechanical interaction, or PCMI, causes a contact pressure between the pellet and the clad. The clad should be able to endure the stresses of PCMI without failure.

The special interest of the RIA research is in high burnup fuel. There are several reasons that make RIA more severe for high burnup fuel. The amount of the fission gases in the fuel increases with increasing burnup. Consequently, swelling becomes more significant with increasing fission gas concentration in the fuel. The diametral clearance for fresh fuel is typically over 100μm. However, the gap in burnt fuel rod may be far less than that or even closed because of the swelling of the pellet and inward creep of the clad during the irradiation. The more extensive swelling and smaller gap, which gives less room for the pellet expansion, result in higher stresses in the clad during PCMI (Bailly et al. 1999).

The clad of a high burnup fuel may be significantly more brittle than in fresh rod, because of the high temperature corrosion and the irradiation hardening in the reactor environment. The high temperature corrosion leads to the formation of an oxide layer on the outer surface of the clad. The thickness of the zirconium oxide or zirconia layer of a high burnup fuel may be up to a hundred microns, which is already a significant portion of the clad total thickness. The oxide layer is very brittle and basically can not withstand tensile stresses. The hydrogen that is released during the oxidation process forms zirconium hydride precipitates in the clad. These precipitates harden the clad while making it more brittle. The hydride precipitates are also preferred places for crack initiation. All of these phenomena decrease the clad ductility and its ability to withstand the high stresses caused by RIA.

The extraordinary conditions of reactor environment in RIA pose great demands on the computer code. The SCANAIR code is designed to simulate the closely interconnected thermal-mechanical behaviour of a fuel rod in RIA transient. The code is especially designed for the simulation of high burnup fuel and it is capable of taking into account many of the high burnup effects described before. The code is developed by the French IPSN research organisation. VTT Energy has contributed to the development work. Recently a creep model was implemented in the SCANAIR code at VTT Energy (Knuutila 2000).

The SCANAIR code

SCANAIR is capable of simulating fuel behaviour in a fast transient, such as RIA. The code is a 1½-dimensional model of a single fuel rod. The code consists of three main modules that are thermal dynamics, structural mechanics, and fission gas behaviour module.

The thermal dynamics module is a finite-volume model of the fuel rod. The thermal module includes models for the fuel rod, the channel, the shroud, and the by-pass. The model calculates only the radial heat exchange of the fuel rod and the shroud. The axial connection is only through the coolant. The heat exchange over the gas gap in the fuel rod is taken into account by calculated gap conductance figure that includes conductive and radiative modes of heat exchange.

The mechanical module is a finite element model of the fuel rod. It includes models for the elastic-plastic deformation of the rod, cracking in the fuel and the zirconia layer of the clad, and fuel swelling. Also a creep model has been recently implemented in the mechanical module.

The fission gas module is used to model the release of the fission gases to the free space of the rod and the swelling due to fission gases in the fuel.

The code must be able to handle the interaction of the wide variety of physical phenomena. This also makes the three modules closely interconnected: The solution of the mechanical module affects the solution of the thermal dynamics module through gap width, which has a strong effect on gap conductivity. Similarly the temperature distribution solved in thermal dynamics module has a strong effect on mechanical properties of the rod materials. The fission gas behaviour is dependent on the temperature distribution and additionally on the stress state of the fuel. The release of fission gases changes the gap conductivity and thus the solution of the thermal dynamics module. The fission gas module is connected to the mechanical module through the swelling of the fission gases in the fuel and the internal pressure of the rod.

The code is also designed to take into account the essential high burnup fuel effects. The code includes models for high burnup effects like the more extensive swelling and the zirconia layer in the clad. If burnt fuel is analysed, SCANAIR needs input from a steady state fuel performance code. The steady state code provides the initial state of the fuel rod characterised by the porosity and burnup of the fuel, fission gas released inside the rod, and fission gas distribution in the fuel and the rod geometry.

Creep

Materials are subject to time-dependent permanent deformation at elevated temperatures even when the stresses are well below the material yield stress. This deformation mechanism is termed as creep. Creep is usually encountered when the temperature exceeds half of the material absolute melting point. The creep mechanisms are closely related to the self-diffusivity of a material. This is why creep rate equations often remind diffusion equations.

Typical material creep behaviour under a constant stress can be divided into three stages that are primary, secondary, and tertiary creep. These stages can be explained by competing hardening and softening mechanisms. At first, the deformation rate decreases, because the hardening mechanisms are in control. At the secondary stage, the hardening and softening mechanisms are in dynamic balance and the deformation rate is constant. This is why the secondary creep is often termed as steady-state creep. At the tertiary stage, material has suffered so much damage that the deformation rate increases and ultimately leads to the failure of the material. A typical high temperature deformation response under a constant stress is represented in figure 52 (Hertzberg 1995).

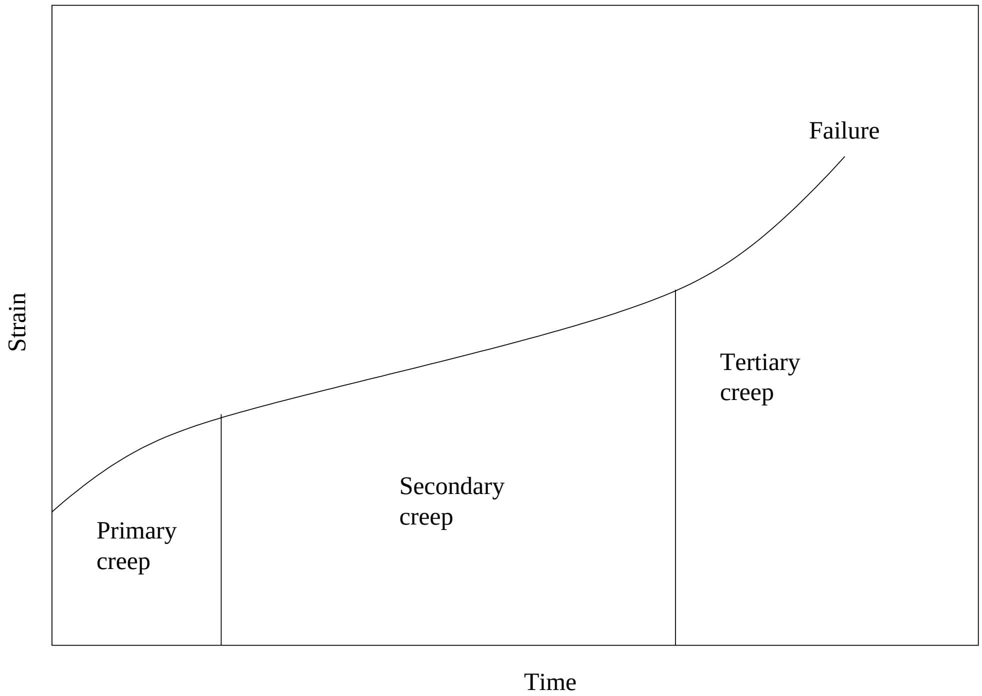

Figure 52. Typical material deformation response at high temperature.

The nuclear fuel, uranium dioxide is a ceramic material and like most of the ceramics, it is hard and brittle at room temperature. At temperatures below half of the absolute melting point (3113K) the behaviour of the uranium dioxide is brittle. The failure is always brittle and no significant plastic deformation can be observed. At higher temperatures, brittle behaviour changes to ductile behaviour. There is a number of factors affecting the creep behaviour of uranium dioxide, such as grain size, porosity, and chemical composition The chosen creep law for the uranium dioxide fuel was found from MATPRO materials database (MATPRO). Steady state creep rate for uranium dioxide is

$$\dot{\varepsilon}_S = \frac{0.3919\sigma \exp(-376900/RT)}{(D-87.7)G^2} + \frac{2.0391x10^{-25}\sigma^{4.5} \exp(-550238/RT)}{D-90.5}, \qquad (18)$$

where $\dot{\varepsilon}_S$ is the creep rate, σ is uniaxial stress, R is the universal gas constant, T is temperature, D is the density (percent of the theoretical density), and G is grain size. The creep rate in transient is evaluated from

$$\dot{\varepsilon}_T = \dot{\varepsilon}_S (2.5 \exp(-1.4x10^{-6} t) + 1), \qquad (19)$$

where $\dot{\varepsilon}_T$ is transient creep rate and t is the time from the start of the transient. The creep law is based on uniaxial compressive creep tests.

The clad is the part of the fuel rod that separates the nuclear fuel from the coolant. The clad of a LWR is made of zirconium alloy. Zircaloy is a zirconium alloy that is used in western PWRs. The creep law proposed by Matsuo was used for the zircaloy. The Matsuo's creep law for zircaloy clad is

$$\begin{aligned} \varepsilon &= \varepsilon_p^s \{1 - \exp[-3120(\dot{\varepsilon}_s t)^{0.5}]\} + 3600\dot{\varepsilon}_s t, \\ \varepsilon_p^s &= 0.0216(\dot{\varepsilon}_s)^{0.109}, \\ \dot{\varepsilon}_s &= 4.3611x10^9 (E/T)[\sinh(1130\sigma/E)]^{2.1} \exp(-272000/RT), \end{aligned} \qquad (20)$$

where $\dot{\varepsilon}_s$ is the steady state creep rate, ε is the creep strain, and E is elastic modulus of zircaloy. The Matsuo's creep law is based on creep tests made with pressurised zircaloy tubes (Matsuo 1987).

Both of the creep laws include the transient primary creep and the steady state secondary creep but they neglect the tertiary creep. However, the tertiary creep stage is very short and a material that has reached it can be considered to be failed.

Creep model in SCANAIR

The uniaxial creep laws of the fuel rod materials presented in the previous section had to be fitted to three-dimensional stress state of the FE model of SCANAIR. The solution to this is very similar to the plasticity models. The von Mises effective stress is used to fit the uniaxial creep law to the actual three-dimensional creep model. The von Mises stress for a stress state consisting of three stress components is

$$\overline{\sigma} = \sqrt{\frac{1}{2}\left[(\sigma_r - \sigma_\theta)^2 + (\sigma_\theta - \sigma_z)^2 + (\sigma_z - \sigma_r)^2\right]} \qquad (21)$$

where σ_r, σ_θ, and σ_z are the stress components in radial, hoop, and axial direction respectively. An associative flow rule is used to evaluate the direction of the creep strain increments. The associative flow rule means that the direction of the creep strain increment is parallel to the stress derivative of the von Mises stress.

The FE model allows to consider the creep strains in certain points called integration or Gauss points. The creep rate in Gauss point can be evaluated using the von Mises effective stress and the temperature of the Gauss point in the uniaxial creep law

$$\dot{\varepsilon} = \dot{\varepsilon}(\overline{\sigma}, T). \tag{22}$$

The effective creep strain increment for the current time step is the creep rate multiplied by the length Δt of the time step

$$\Delta\overline{\varepsilon}_c = \Delta t \dot{\varepsilon}. \tag{23}$$

The actual strain state in the SCANAIR Gauss point is three-dimensional. The effective value of the creep strain increment has to be now fitted to the three dimensional strain state of SCANAIR. This is done using an associative flow rule. The flow rule sets the direction of the creep strain increments. The associative flow rule means that the direction of the plastic strain increment is perpendicular to the curve, in which the von Mises effective stress has a constant value. The direction of the creep strain increment is thus parallel to the stress derivative of the von Mises effective stress at the current Gauss point. The creep strain increment is

$$\Delta\varepsilon_c = \Delta t \dot{\varepsilon} \frac{\partial \overline{\sigma}}{\partial \sigma} \tag{24}$$

One has to still decide which stress state is used to evaluate the strain rate and the flow rule. An explicit integration of the creep uses the stress state at the beginning of the current time step. An implicit integration scheme uses the stress state at the end of the current time step. The stress state at the end of the current time step is unknown and it has to be iterated. The explicit scheme is very simple and it seems very tempting. However, the explicit integration of the creep needs very small time steps when the creep rate is high, as it often is in the fast RIA transient. It was concluded that the explicit integration of the creep is inadequate for the fast loading that is encountered in RIA. An implicit scheme that uses Newton-Rhapson iteration with a line search to find the creep strain increments in the Gauss points was implemented in SCANAIR. The advantage of the implicit integration of the creep is its fast convergence and unconditional stability allowing large time steps.

RIA calculation

A simple test case was run to discover the effect of the creep model in RIA calculation. A power pulse of 20ms in length and with peak fuel enthalpy of 590J/gUO2 was calculated with SCANAIR. The fuel rod for the test calculation is similar to the REP-Na 4 test rod. The REP-Na tests are RIA tests that were made in the sodium cooled CABRI test reactor in France. The maximum burnup of the rod is 62 MWd/kgU. The coolant in the test case is sodium like in REP-Na tests. Only the power pulse was modified in the REP-Na 4 test case, because with the actual REP-Na power pulse the fuel temperature stays so low that no significant creep is observed. A hypothetical test case with a more energetic power pulse was calculated. Figure 53 shows the radial temperature distributions during the power pulse. Figure 54 shows the radial distribution of the effective creep strains in the fuel rod. Figure 55 shows the radial distribution of the effective plastic strains in the fuel.

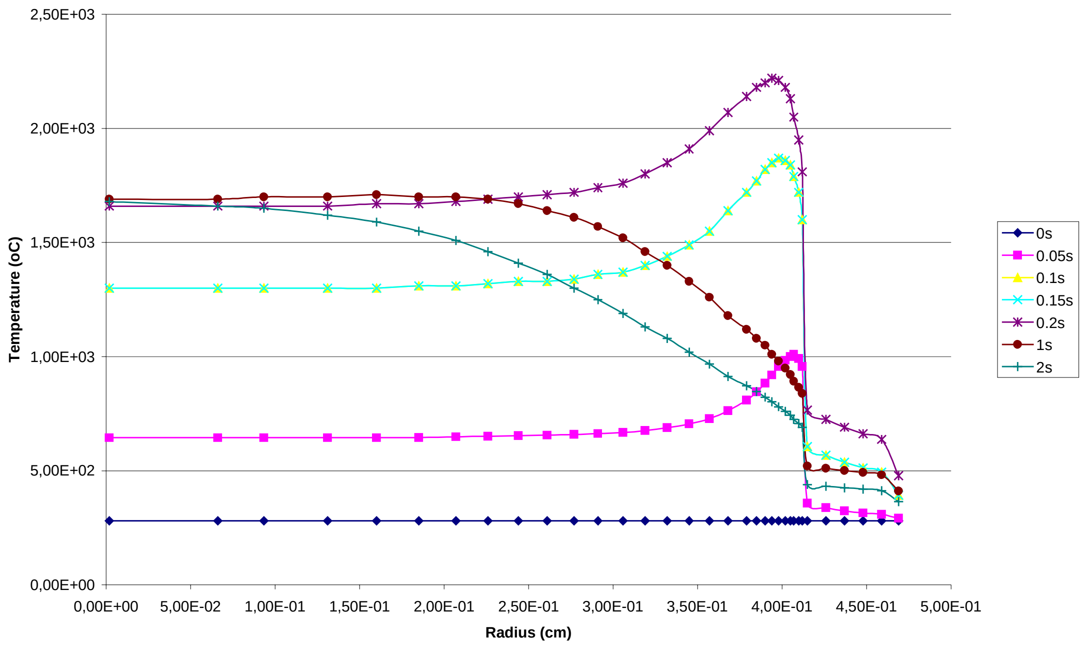

Figure 53. The radial temperature distributions of the fuel rod during the power pulse.

The test case clearly shows that the creep is an important deformation mechanism in the fuel and it cannot be neglected in RIA calculations. When the fuel temperature exceeds 2000°C the creep becomes controlling deformation mechanism over the plastic deformation. However, the clad stays so cool that the plastic deformation is the controlling deformation mechanism in it. The creep would be even more important deformation mechanism if less effective cooling should be used, such as water.

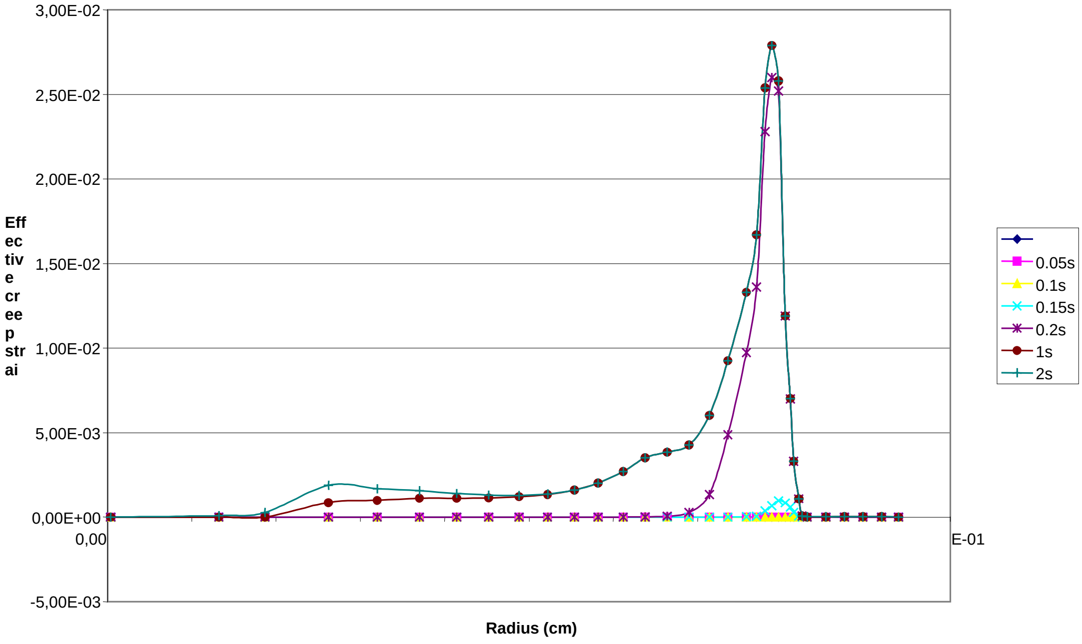

Figure 54. The radial distribution of the effective creep strains in the fuel rod.

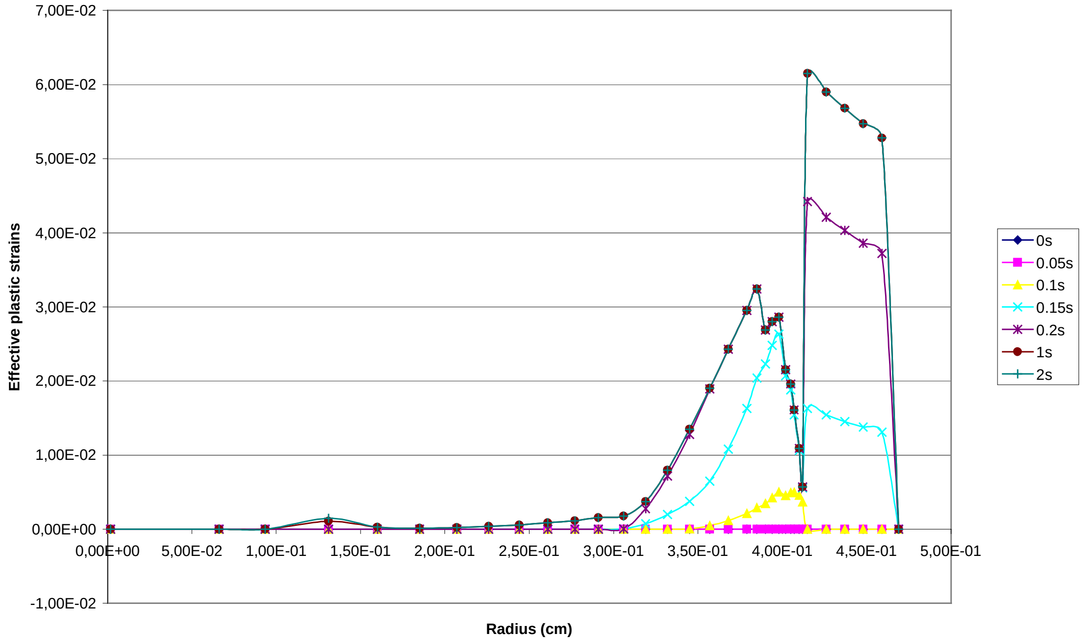

Figure 55. The radial distribution of the effective plastic strains in the fuel rod.

Conclusions

The high stresses and strain rates in loading caused by RIA pose great demands on a creep model. Only implicit integration of the creep gives the robustness needed in RIA calculation. The advantage of the implicit integration is that it allows long time steps and posesses unconditional stability.

The creep model is needed to simulate the softening of the fuel at elevated temperatures. The plasticity model is not adequate for this because it can not describe the stress relaxation in the fuel. The test calculation showed that the creep becomes an important deformation mechanism in RIA when the fuel temperature exceeds 2000°C.

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5.6 READY special report 1

Application of TRAB-3D to BWR and PWR transient calculations

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Introduction

TRAB-3D is the newest member in the reactor dynamics code family developed in VTT. It is now possible to analyze both PWRs and BWRs taking into account the real core loading. The core geometry can be either rectangular, as in most western reactors, or hexagonal, as in e.g. VVER reactors. In addition to the core models the codes include models for the cooling circuits and reactor control systems. This paper describes the neutronics/thermal-hydraulics coupling methodology and validation of the three-dimensional TRAB-3D for the application to PWR and BWR transient calculations.

Code development in the fields of reactor physics and dynamics, as well as in thermal-hydraulics has been one of the key areas of reactor safety research in Finland since the middle of the seventies. In reactor dynamics, which combines neutron kinetics, heat conduction, heat transfer and hydraulics of the cooling circuit, the first product was a LWR core model (Rajamäki 1980a), which is basically one-dimensional two-group code, but includes a synthesis model for radially nonuniform dynamics.

The same core model was applied in the development of the BWR dynamics code TRAB (Rajamäki 1980b and Räty et al. 1991), which is continuously used in the safety analysis of the Finnish TVO power plant. It models the core and the main circulation system inside the reactor vessel, including the steam dome with related systems, steam lines, recirculation pumps, incoming and outgoing flows, as well as control and protection systems. Typical applications are the pump trip and the steam line isolation. The code has also been used for the simulation of the RBMK type reactor in accident conditions.

The HEXTRAN (Kyrki-Rajamäki 1995) core dynamics model has been developed on the basis of the stationary two-group diffusion code HEXBU-3D (Kaloinen et al. 1981) for hexagonal geometries, with the fuel heat conduction and channel hydraulics description of the TRAB code.

TRAB-3D (Kaloinen & Kyrki-Rajamäki 1997) is a similar code used for rectangular lattice reactors. In addition to the core dynamics TRAB-3D also includes the BWR circuit models of TRAB. The validation work of TRAB-3D consists of the successful calculation of OECD benchmarks, the most recent of which has been the PWR main steam line break (MSLB) benchmark (Ivanov et al. 1999), as well as comparisons with measurements of real TVO plant transients (Daavittila et al. 2000).

As the cooling circuit model of PWRs serves the SMABRE code (Miettinen 1985) coupled with HEXTRAN or TRAB-3D. SMABRE is a fast running simulation code and it was developed for such thermal-hydraulic accidents as small breaks, originally in order to facilitate parameter variations to support slow RELAP calculations. Its range of application was, however, soon extended to cover such types of accidents as small break LOCA, primary-secondary leak, steam line break, ATWS and most parts of a large break LOCA in VVER plants.

Coupling of neutronics and thermal hydraulics

There is a strong coupling between neutronics, fuel heat transfer, and thermal hydraulics in the reactor core. In reactor dynamics codes iterative simultaneous solutions are used for equations describing these different physical phenomena. The coupling is especially strong when boiling is occurring in the core.

In reactor dynamics applications also the combination of neutron-physical phenomena with thermal-hydraulics of the whole cooling circuit is of vital importance. It has therefore been a nearly built-in property of the Finnish reactor dynamics codes, or in the combinations of core models with the circuit thermal-hydraulics models, from the first applications to the present day 3D models. Two different solutions have been used to perform the coupled calculations with cooling circuits. In BWR models (TRAB-3D) the coolant circuit has been included as an essential part together with the core flow equation solution. The neutronics, fuel heat transfer and thermal hydraulics are iterated together for each time-step to achieve a stable solution.

In PWRs the circuit thermal hydraulics is been calculated with a separate model. The combination codes used for square and hexagonal lattice cores with a PWR circuit are TRAB-3D(core model only)/SMABRE and HEXTRAN/SMABRE, respectively. The solution method of the SMABRE circuit model is non-iterative and there is only a loose coupling between it and the TRAB-3D (or HEXTRAN) core model; the codes are connected by data changing once during a time-step, no iterations are made with the circuit hydraulics.

The coupled code TRAB-3D/SMABRE has its own main program and a few interfacing subprograms, but as a rule TRAB-3D and SMABRE are used in the same way as the separate codes. Both codes use their own input, output, restart and plotting capabilities. TRAB-3D dictates the time step chooses.

SMABRE solves the thermal hydraulics both in the cooling circuits and in the core, in addition to this it has an own model for the core, based on the point kinetic solution and own fuel heat structures which can be used as an alternative of the 3D calculation. For SMABRE typically a few core axial nodes and radial sectors are defined (e.g. 6 sectors and 5 axial nodes for a VVER-440); the structure of this SMABRE core model is dictated by the pressure balance calculation of the whole primary system. In order to accurately calculate the fission power behaviour with feedback effects TRAB-3D (or HEXTRAN) solves both the thermal hydraulics and neutronics of the refined core model, with about 20-25 axial nodes and each fuel element calculated separately with an own flow channel. The 1D radial fuel heat transfer in the pellet, gas gap and cladding for average rod in each assembly is calculated with typically 8 radial nodes.

The interchanged quantities between the modules are the power to coolant distribution from TRAB-3D (or HEXTRAN) to SMABRE and the core outlet pressure, total core inlet mass flow, inlet mass flow enthalpies and boron concentrations of each core sector from SMABRE to TRAB-3D (or HEXTRAN). No more data changes are needed. This type of coupling, the parallel coupling, has shown to be stabile and effective and it was used already in the coupling between the axially one-dimensional core model TRAB and SMABRE.

Application to PWR: OECD main steam line break (MSLB) benchmark

The purpose of the OECD PWR MSLB benchmark is to test the coupling of 3D core neutronics models to plant thermal hydraulics models. The reference problem is a rupture of one steam line upstream of the main steam isolation valves at the Three Mile Island Unit 1 (TMI-1) NPP. The plant has two circulation loops, thus a steam line break in one loop leads to strongly asymmetric cooling, which makes the case interesting for 3D dynamics codes. Additionally one control rod is assumed stuck in the following reactor trip.

The benchmark has been calculated in three stages: 1) A plant simulation with point kinetics, 2) 3D neutronics/core thermal hydraulics simulation and 3) Coupled 3D neutronics/plant thermal hydraulics calculation. At VTT Energy all the stages were successfully calculated with SMABRE, TRAB-3D core model, and the coupled code TRAB-3D/SMABRE, respectively.

The TRAB-3D core model of the TMI-1 plant consists of 177 hydraulic channels, one for each fuel bundle, which are divided in 24 axial nodes. For the third stage of the benchmark this model was coupled to the SMABRE models for the primary loops, the secondary loops, and the break itself. Because in the coupling scheme described in the previous chapter, the SMABRE loops are closed, a parallel hydraulic model of six sectors for the core was included, in addition to the TRAB-3D core hydraulics.

Using the best-estimate cross sections there is no power peak after the reactor trip with the 3D kinetics models. To make comparisons between codes more interesting, a scenario with reduced control rod worths for the trip was specified. The time behaviour of power for this second scenario is shown in figure 56. Figure 57 shows the radial power distribution at the time of the power maximum after the reactor trip. The effect of the asymmetric cooling is clearly visible. In the core half on the side of the broken secondary loop, the power level is significantly higher than on the side of the intact secondary loop. Further, the power generation is strongly concentrated near the position of the stuck control rod. Overall, the results with TRAB-3D/SMABRE are well in agreement with the results from the other participants.

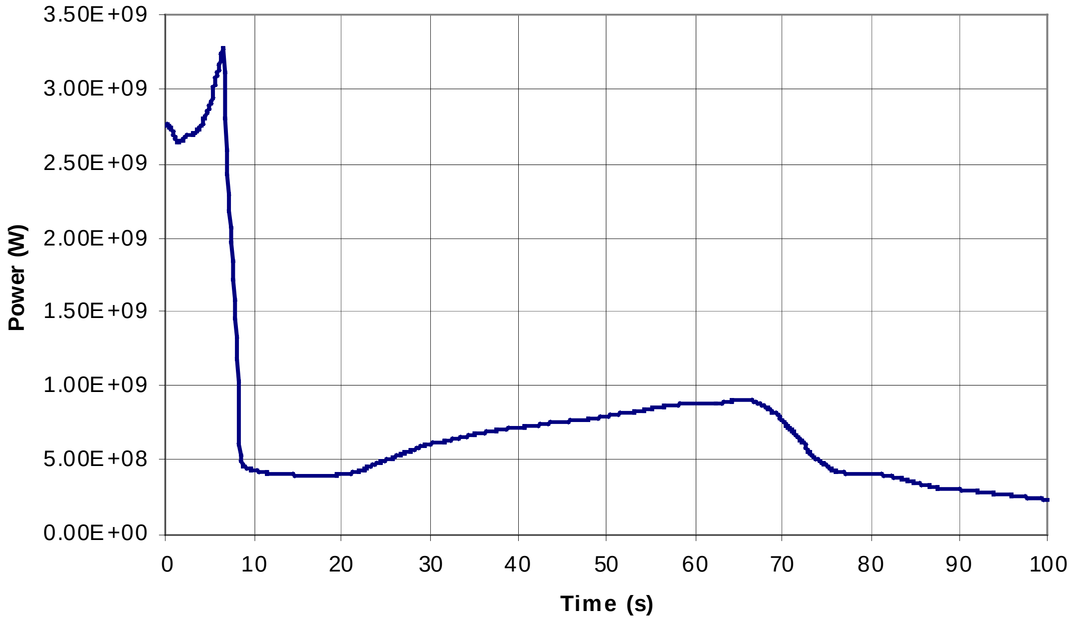

Figure 56. Reactor power vs. time, TRAB-3D/SMABRE calculation of the OECD PWR MSLB benchmark.

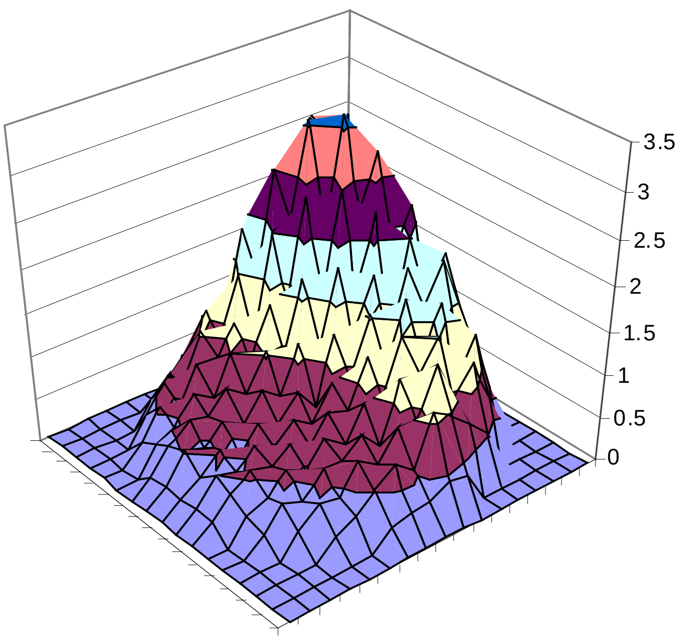

Figure 57. Radial power distribution at the time of power maximum after reactor trip.

Application to BWR 1: Olkiluoto 1 overpressurization transient

The core overpressurization transient occured in September 10, 1985 in the Olkiluoto 1 reactor. The transient was caused by erroneous functioning of the pressure controller, which led to the closing of the turbine valves in about 0.5 seconds. The pressure of the reactor increased to 78.5 bar, which in turn led to a decrease in the core void fraction and an increase in reactor power. The transient was terminated safely with all safety systems functioning normally except for one relief valve that did not open.

In the beginning of the transient the reactor was operating at full power (2160 MW, 100%) with a mass flow rate of 7248 kg/s and system pressure 70 bar. The core loading during the event was highly mixed. A number of burnt bundles from the initial core were reloaded in the beginning of the cycle and at the same time the batches of fresh fuel were changed from ABB bundles to KWU fuel. Thus a large number of different sets of homogenized cross sections had to be developed for neutronic calculations of these cycles with the CASMO-4 code. There is also a variation in the thermal hydraulic characteristics of the fuel channels since the core loading includes bundles of both 8x8 and 9x9 rod lattice. All this makes the analysis of these cycles a good test for a three-dimensional dynamics code.

The reactor core was described by a half core symmetry sector with periodic boundary conditions. One hydraulic channel and one neutronic node in a horizontal cross section was associated with each of the 250 fuel bundles in a half core sector. Axially the bundles were divided into 25 hydraulic and neutronic nodes of equal height, which is the standard division used in In-Core-Fuel Management (ICFM) calculations of ABB (Westinghouse) Atom reactors.

For thermal hydraulic calculations the fuel bundles were described with four groups of different input specifications in hydraulics and heat transfer. The groups consist of ABB fuel with original or new spacers and KWU fuel with 8x8 or 9x9 lattice, situated in three zones of different inlet orifice pressure loss.

The nodal distributions of burnup and void and control rod histories were obtained from ICFM calculations carried out with POLCA-4. Also radial and axial reflectors of the core were described with the same albedo boundary conditions as in ICFM calculations.

The time histories of reactor power and system pressure from the TRAB-3D simulations of the transient are shown together with the measured values in figure 58 and 59. The calculated fission power agrees well with the measurement data, but unfortunately the measurement system could not register power levels greater than 130%. The calculated maximum power is 5737 MW (266% of nominal). The time behaviour is very close to the measurement.

The maximum pressure calculated by TRAB-3D is 79.1 bar, which is less than one bar greater than the measured maximum. The calculated time of the maximum differs a little from the measured one. According to the measurements the first local pressure maximum is the global maximum, whereas in the TRAB-3D results the maximum pressure during the transient occurs at the time of the second local maximum. Variation calculations have shown that the results are sensitive to such uncertainly known parameters as gas gap conductance, evaporation model and carry under of steam in the downcomer. The timing of local pressure maxima could be better forecasted by modelling individually all the four steam lines. It would be possible to include them in the present model.

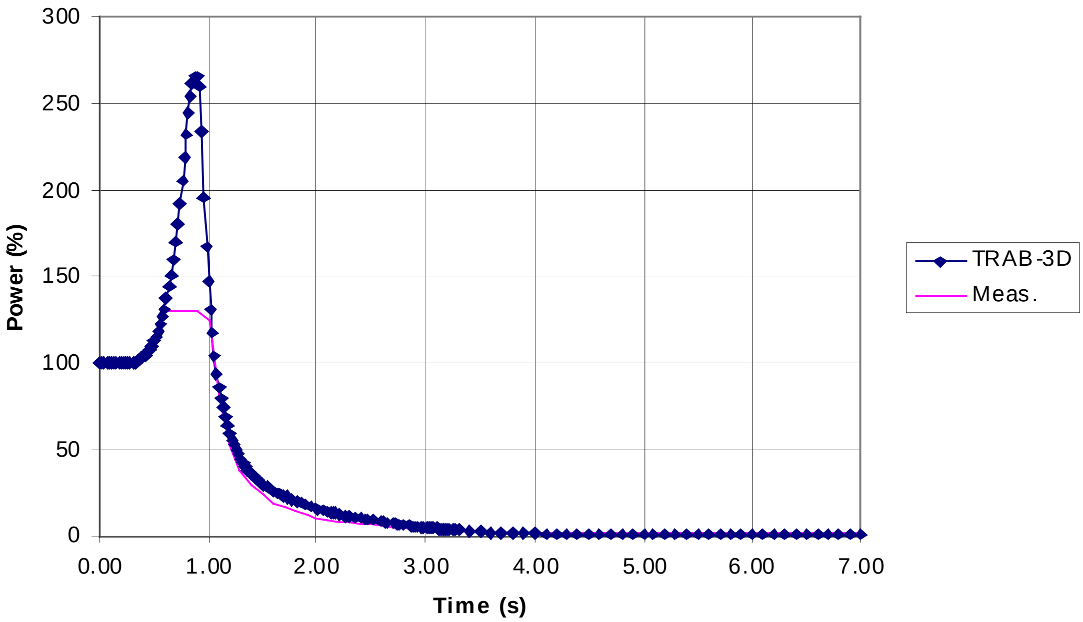

Figure 58. Calculated fission power against the measured values.

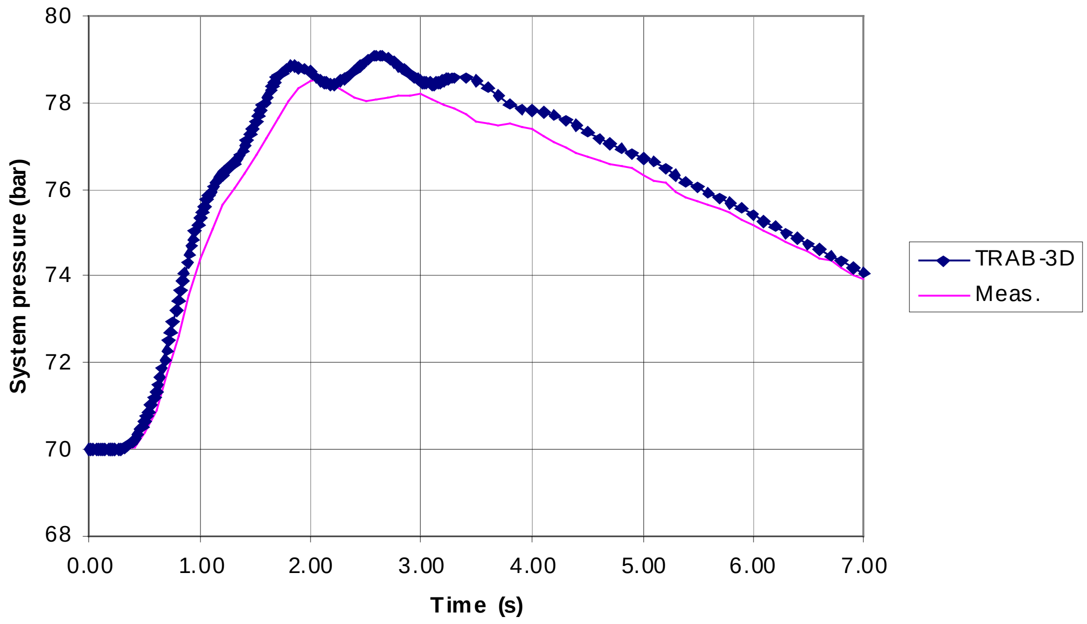

Figure 59. Calculated system pressure against the measured values.

Application to BWR 2: Olkiluoto 1 oscillation incident

The oscillation incident took place in the Olkiluoto 1 reactor in February 22, 1987 during the reactor startup procedure. The restart was carried out by a control rod withdrawal with the recirculation pumps at the minimum speed. When a 60% power level was reached, with a 30% mass flow rate of the nominal value, the fission power started to oscillate. At this point a periodical test of the by-pass valves of one feedwater preheater was performed. The reconnection of the preheater failed resulting in a 40 K drop in the feedwater temperature. When the cold water reached the reactor core the oscillation started to diverge. After a few seconds the amplitude of the oscillation was sufficient to cause a reactor scram which terminated the incident. The measured fission power during the incident is shown in figure 60.

According to the utility the reactor power and mass flow rate in the beginning of the oscillation incident were 1298 MW and 2980 kg/s, respectively. According to the TRAB-3D calculation this is a stable state, even with the feedwater temperature disturbance. The reactor state can be easily destabilized by adding a local pressure loss to the region of two-phase flow. A small additional pressure loss at the end of the core channels (one tenth of the local pressure loss at the core inlet) is sufficient to make the reactor unstable in the above conditions.

In the present calculations, however, the reactor model was kept the same as in the overpressurization transient simulations. The reported mass flow rate includes some uncertainty, due to the unclear initial state. Decreasing the mass flow rate to 2500 kg/s makes the reactor unstable. Another possibility is to increase the power level. If the fission power in the initial state is increased to 1500 MW the reactor becomes, again, unstable. For the initial state of the dynamic calculations the power level was increased to 1400 MW and the mass flow rate reduced to 2650 kg/s. This state is close to the stability limit and there is a small-amplitude limit-cycle oscillation, which starts to diverge after the feedwater disturbance.

Figure 61 shows the calculated fission power for the oscillation incident, which is terminated by reactor scram. The overall behavior is similar to the measured fission power of figure 60. The oscillation frequency, 0.45 Hz, is close to the measured frequency 0.42 Hz and the beat, caused by the interaction of the core channels and the by-pass channel is clearly present in both figures. The observed oscillation frequency could probably be obtained by varying the gas gap conductance, which has a direct effect on the time constant of the heat transfer.

The validation of a dynamics code using an oscillation incident is not as straightforward as in the case of a more definite transient that starts from zero or full power. It is not possible to reproduce every oscillation cycle exactly the same as in the measurement data. It is, however, possible to check if all relevant phenomena are present in the simulated results and to compare parameters such as the oscillation frequency. The stability of a given state is highly sensitive to small changes in the plant model, for example the local pressure losses and the geometry of the core by-pass channel have a significant effect on the core stability. Therefore an analysis of this kind of incident is fairly difficult.

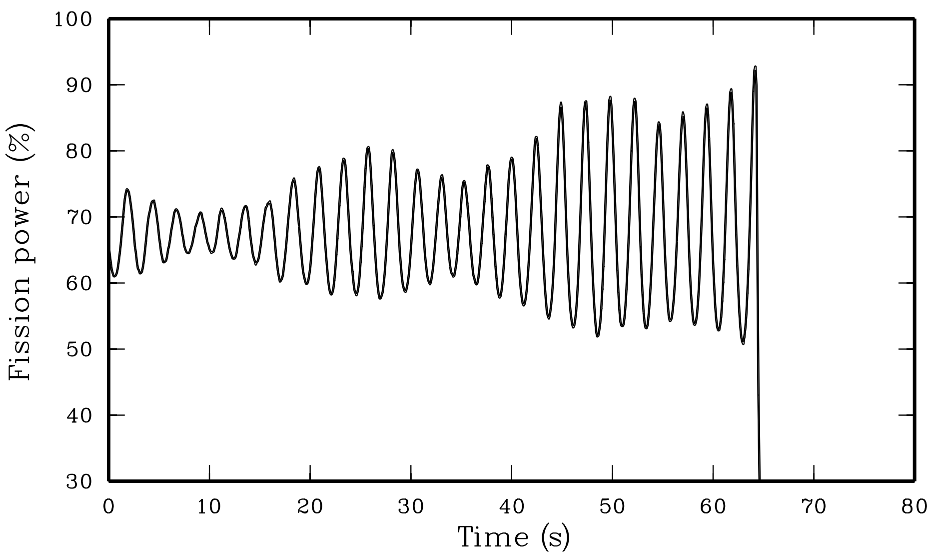

Figure 60. Measured power during the oscillation incident.

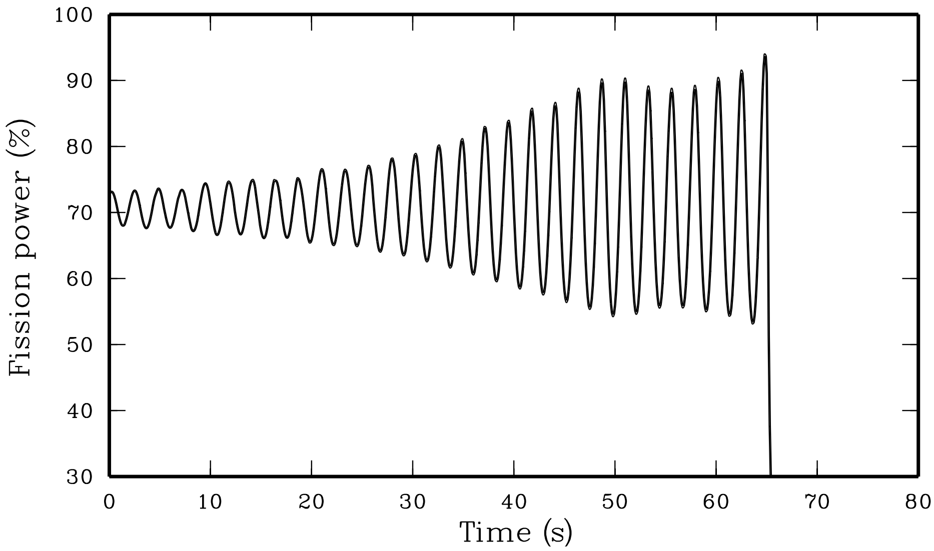

Figure 61. Calculated power during the oscillation incident.

Conclusions

In Finland a number of transient and accident analysis codes have been developed during the past twenty years mainly for the needs of our own power plants, but some of the developed methods have also been utilized elsewhere. In these codes the strong coupling between neutronics and thermal hydraulics in the reactor core has been solved by using iterative simultaneous solutions of equations describing these different physical phenomena. Two different solutions have been used to perform the coupled calculations with cooling circuits. In BWR models the coolant circuit has been included as essential part together with the core flow equation solution. In PWRs (or VVERs) the circuits have been calculated with a separate model connecting with the core model only by data changing once during a time-step. Both solutions have been applied successfully as shown again by the examples here.

The validation cases of TRAB-3D against real plant measurements and international benchmark calculations with other codes show that the code is a valuable tool to be applied in BWR and PWR transient and accident calculations.

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5.7 READY special report 2

Three-dimensional analytic function expansion nodal model for the TRAB-3D nodal code

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Introduction

Static and dynamic analyses of nuclear reactors are usually carried out with diffusion-theory based *nodal methods*, in which the fuel assemblies are modelled in the horizontal direction as one or four and in the axial direction as 10–30 homogenized volumes. The homogenized two-group cross sections needed in the nodal analysis are obtained from accurate single-node heterogeneous multigroup transport calculations.

The nodal models differ from each other in the way the intra-nodal flux shape is approximated in calculation of the coupling coefficients between adjacent nodes. The simplest methods use polynomials or other simple functions for direct interpolation of the fast and thermal fluxes; a more advanced approach, used e.g. in the HEXBU and TRAB-3D nodal models (Kaloinen & Kyrki-Rajamäki 1997), is to decompose the two-group flux into eigenmodes, which have a better-defined shape than the fast and thermal flux and can thus be interpolated more accurately.

Especially in cores with large flux variations between adjacent nodes, it is important to obtain as accurate an estimate of the intra-nodal flux shape as possible. Another need for improving the flux shape comes from pin power reconstruction. Most of the current nodal codes (including TRAB-3D and HEXBU) do not, however, use corner point flux values in the nodal calculations and information about the flux near node corners is therefore limited. A new nodal model, based on the *analytic function expansion nodal model* (AFEN), initially developed in South Korea (Noh & Cho 1994), has been developed for calculating the three-dimensional neutron flux. In addition to improved accuracy within the core, the analytic function expansion enables also calculation of the flux in reflector areas,[1] thus possibly enabling better evaluation of e.g. the flux seen by ex-core detectors. This paper briefly describes the model and the results of the first test calculations.

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[1] Within the limitations of the diffusion theory.

Nodal Model

AFEN method uses the flux eigenmodes instead of the two-group fluxes. The eigenmodes and the corresponding modal bucklings are obtained by substituting the mode functions Ψ satisfying Helmholz equation

$$\nabla^2\Psi - B^2\Psi = 0 \tag{25}$$

and

$$\begin{aligned} \Phi_1 &= \Psi_1 + R_I \Psi_2 \\ \Phi_2 &= R_{II}\Psi_1 + \Psi_2 \end{aligned} \tag{26}$$

into the two-group diffusion equation

$$\begin{bmatrix} -D_1\nabla^2 + \Sigma_{a1} + \Sigma_{12} - \frac{1}{k_{eff}}\left(\nu\Sigma_f\right)_1 & -\frac{1}{k_{eff}}\left(\nu\Sigma_f\right)_2 \\ -\Sigma_{12} & -D_2\nabla^2 + \Sigma_{a2} \end{bmatrix} \begin{bmatrix} \Phi_1 \\ \Phi_2 \end{bmatrix} = 0 \ . \tag{27}$$

The two-group diffusion equation is thus replaced with the simpler Helmholz equations (25), with the characteristic (or modal) bucklings $B^2_{I,II}$ containing all information about cross sections and other diffusion parametres.

The AFEN method is based on interpolating the modes with form functions satisfying the Helmholz equation. Depending on whether the buckling is positive or negative, these are either hyperbolic or trigonometric sine and cosine functions, respectively. If the flux is assumed separable in the axial and radial directions, this leads to form functions of the form

$$\begin{aligned} \Psi(x,y,z) = & \left(\sum_{n=1}^{8} \left(\tilde{a}_n CS\left(b_{xy}\left(\alpha_n x + \beta_n y\right)\right) + \tilde{b}_n SN\left(b_{xy}\left(\alpha_n x + \beta_n y\right)\right)\right) + C \right) \\ & \left(c_1 CS\left(b_z z\right) + c_2 SN\left(b_z z\right)\right) \quad , \end{aligned} \tag{28}$$

where b is the square root of the absolute value of buckling eigenvalue and *CS* and *SN* either the hyperbolic or trigonometric sine and cosine functions depending on whether b is positive or negative, respectively. The sets (α_n, β_n) consist of permutations of $(0, \pm 1)$

and $\left(\pm\frac{1}{\sqrt{2}},\pm\frac{1}{\sqrt{2}}\right)$. The constant term forces the nodal balance equation to be satisfied in the early stages of iteration, when the effective multiplication factor has not yet converged. This term will vanish during the course of iteration.[2]

Using the relations

$$\begin{aligned}
\sin(x+y) &= \sin(x)\cos(y)+\cos(x)\sin(y)\\
\cos(x+y) &= \cos(x)\cos(y)-\sin(x)\sin(y)\\
\sinh(x+y) &= \sinh(x)\cosh(y)+\cosh(x)\sinh(y)\\
\cosh(x+y) &= \cosh(x)\cosh(y)+\sinh(x)\sinh(y)
\end{aligned}\quad, \tag{29}$$

the radial component of (28) can be rewritten as

$$\begin{aligned}
\Psi(x,y) = \ & a_1 SN(bx)+a_2 CS(bx)+a_3 SN(by)+a_4 CS(by)\\
+\ & b_1 SN\left(\frac{\sqrt{2}}{2}bx\right)SN\left(\frac{\sqrt{2}}{2}by\right)+b_2 SN\left(\frac{\sqrt{2}}{2}bx\right)CS\left(\frac{\sqrt{2}}{2}by\right)\\
+\ & b_3 CS\left(\frac{\sqrt{2}}{2}bx\right)SN\left(\frac{\sqrt{2}}{2}by\right)+b_4 CS\left(\frac{\sqrt{2}}{2}bx\right)CS\left(\frac{\sqrt{2}}{2}by\right)\\
+\ & C \quad ,
\end{aligned}\tag{30}$$

which is more convenient for further derivations, as each term is either symmetric or antisymmetric with respect to *x* and *y*.

Coupling Equations

In order to derive the coupling equations between the nodes, a set of nodal unknowns must be chosen. Since there are 9 undetermined coefficients in the *xy*-plane flux representation (28), nine nodal unknowns are required. The natural choice for the nodal

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[2] The constant term will vanish when the iteration converges, because there is no such term in the solution of the differential diffusion equation $-\nabla^2\Phi+\Sigma_a\Phi=\frac{1}{K}\nu\Sigma_f\Phi$, from which the buckling is obtained as an eigenvalue solution, and because the converged solution satisfies the diffusion equation both in the differential and integral (nodal balance equation) form. The only contribution of the constant is to force the nodal balance equation to be satisfied in the early iteration steps, when the differential diffusion equation is not yet converged; it has no effect on the other continuity equations written between the currents, as the constant vanishes in differentiation.

unknowns is the node averaged flux, four side averaged fluxes and four corner point fluxes, as shown in figure 62.

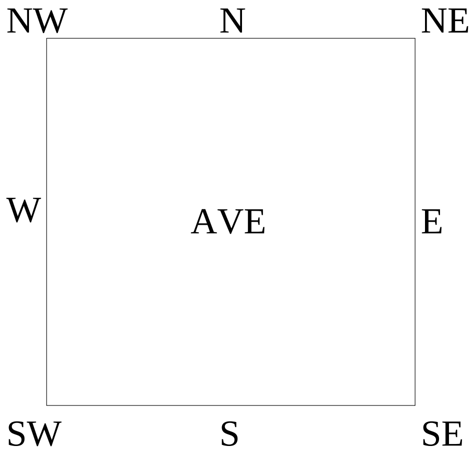

Figure 62. Nodal unknowns in the xy-plane

To couple the nodes, eleven relations (9 in the radial, 2 in the axial direction) are needed. They were chosen as follows: current continuity on each node face (6 equations), corner point balance equations requiring that the vertical node edges do not have neutron sources (4 equations) and nodal balance equation requiring that the leakage of neutrons over node boundaries equals to the difference between neutron sources and sinks within the node (1 equation). This choice of nodal unknowns and continuity equations differs from that made in TRAB-3D in that the corner point fluxes (and, accordingly, corner point balance equations) are included directly in the model.

The coupling coefficients are obtained by solving the expressions of the nodal unknowns for the mixing factors $a_1 \ldots c_2$, and by substituting these solutions into the expressions of boundary currents and corner point currents. The mixing factors are thus not calculated during the iteration; rather, the currents can be expressed directly in terms of the nodal unknowns.[3] Since the continuity equations are for the physical group currents, whereas the coupling coefficients are for flux eigenmodes, a similarity transform of the form

$$\begin{bmatrix} F_{11} & F_{21} \\ F_{12} & F_{22} \end{bmatrix} = -\begin{bmatrix} D_1 & \\ & D_2 \end{bmatrix} \begin{bmatrix} 1 & R_I \\ R_{II} & 1 \end{bmatrix} \begin{bmatrix} F_1 & \\ & F_2 \end{bmatrix} \frac{1}{R_I R_{II} - 1} \begin{bmatrix} -1 & R_I \\ R_{II} & -1 \end{bmatrix} \qquad (31)$$

[3] After the iteration has converged, the mixing factors can be evaluated for pin power reconstruction.

must be performed in order to operate on the physical group fluxes in spite of the eigenmode flux model. In other words: all information about the flux shape within the node is contained in the four *F* factors showing the contribution of each flux component to the currents used in the continuity equations. To get all required currents (total outward current, side-averaged currents and corner point currents), 15 such *F* matrices are needed per node. Since evaluation of the matrices is time-consuming and the flux shape does not change as rapidly as the amplitude, the *F* matrices are not updated on every iteration step.

Test calculations

First test calculations with the new flux model were made on the two-dimensional IAEA-2D benchmark problem (ANL 1977). The results were very similar to those reported by Noh & Cho (1994) in their first published paper on the AFEN method. The results were not identical – the largest differences between assembly-averaged fluxes were of the order of 0.2% – even though the method is in principle the same. The solution of a benchmark problem with given cross sections and no feedback effects is quite sensitive to boundary conditions, and a slight change in the way the reflector is modelled is a plausible explanation for the discrepancies, since the largest deviations were found in the vicinity of the reflector.

The second test problem was obtained by extending the IAEA-2D benchmark in the axial direction to 11 nodes and adding two axial reflector nodes at each end of the reactor (figure 63). In order to get a more detailed view of the flux shapes, a pin power reconstruction module[4] was added to the model. Figure 64 shows the thermal peak in one reflector node (assembly 51 of figure 2) and the effect of a control rod (assembly 70). The fast flux in the same 3×3 node area of the reactor is shown in figure 65. Qualitatively, the results seem correct; however, no comparative calculations have yet been made with the 3D model.

[4] In pin power reconstruction, the mixing coefficients $a_1 ... c_2$ of the flux form functions are evaluated, and the homogeneous flux can thus be interpolated within the node. This "homogeneous power" can then be combined with the heterogeneous single-assembly power distributions in order to get estimates for the powers of individual fuel rods within the reactor. See e.g. [4].

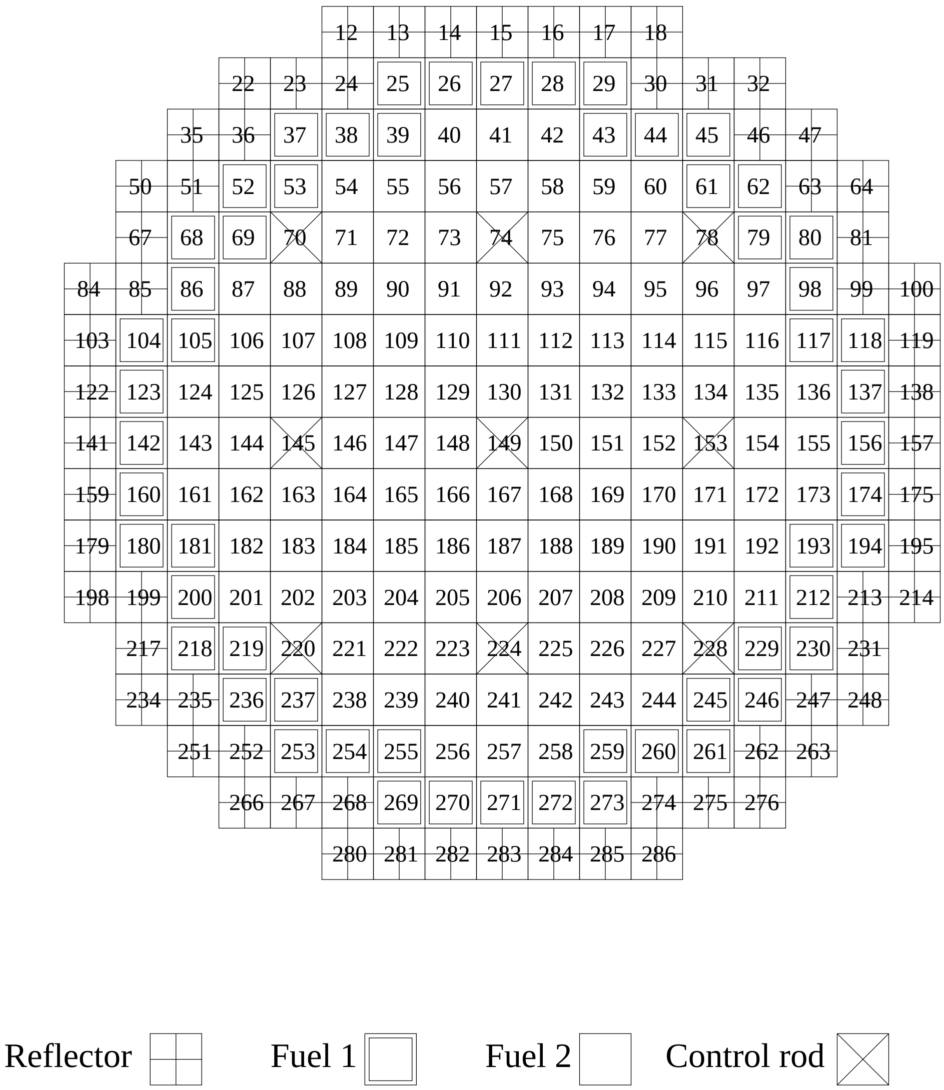

Figure 63. Fuel assembly layout in the IAEA-2D benchmark problem.

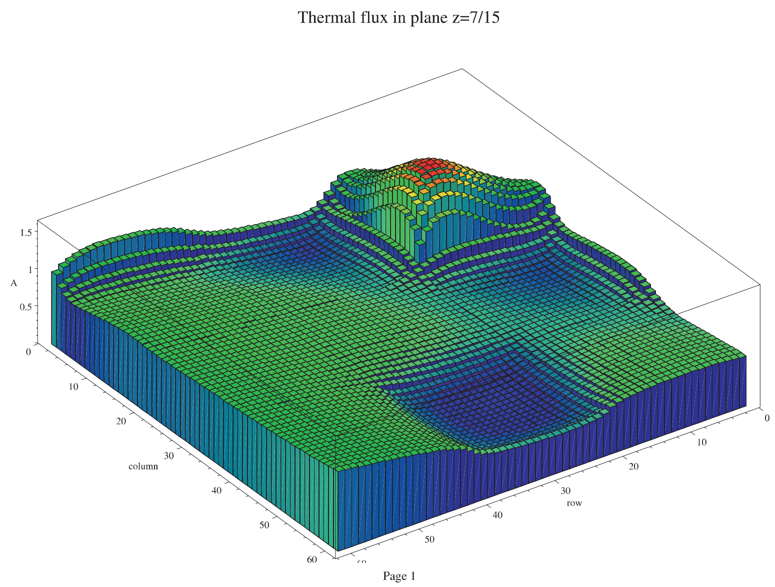

Figure 64. Thermal flux in the vicinity of reflector and a control rod.

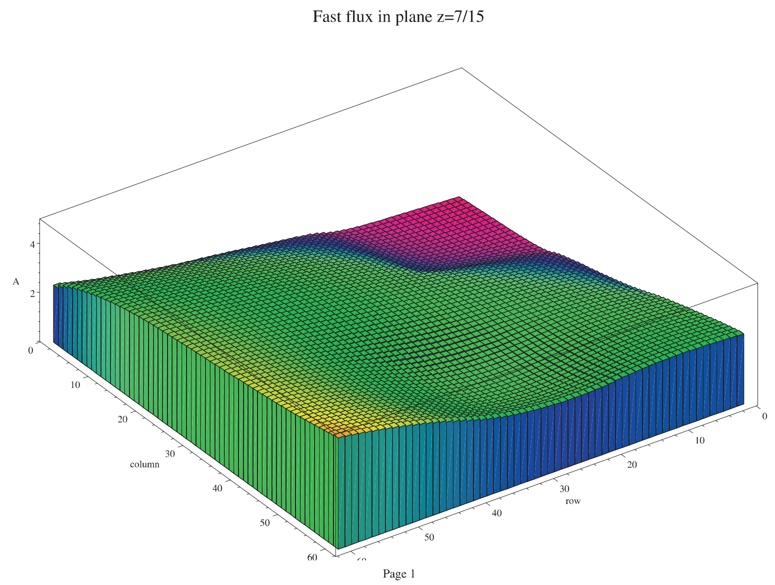

Figure 65. Fast flux in the vicinity of reflector and a control rod.

Future Plans

The next step in the development of the model (after making some comparative calculations on a suitable 3D benchmark problem) is to couple it to the steady-state hydraulics of TRAB-3D. This would enable further comparative calculations on realistic reactor cores.

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5.8 TOKE special report

Modelling of PACTEL with APROS5 simulation environment

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Abstract

PACTEL facility has been used quite extensively for validation of the thermal hydraulic part of APROS Multifunctional Simulation Environment. During the years, several models of PACTEL facility have been created with APROS. The driving forces for creation of new models have been improvements in APROS code models, development of graphical user interfaces of APROS environment and changes in the PACTEL facility. The specific features of the new model described in this paper are that the present PACTEL facility has been modelled in a more detailed manner than in the previous ones, and the model has been created using the new graphical user interface GRADES. The current paper describes the model created in a Master's Thesis work and

preliminary validation calculations made with the model. The paper contains also a short overview on the various PACTEL models created with APROS and the validation calculations performed.

Introduction

The PACTEL test facility at Lappeenranta University of Technology is an out-of-pile experimental facility designed to simulate the major components of the primary loop of a commercial PWR during postulated small and medium-size LOCA's, natural circulation and operational transients. The reference plant is VVER-440 PWR in Loviisa, Finland. The original design of the facility has been described in detail in (Riikonen 1988) and the current facility in (Tuunanen et al. 1998). Experimental results obtained with PACTEL have been used quite extensively for the validation of thermal hydraulic part of the APROS Multifunctional Simulation Environment (Silvennoinen et al. 1989). PACTEL results have been used quite extensively also in validation of RELAP5 computer code at VTT Energy, and also in validation of other computer codes, in particular in context of the international ISP-33 benchmark.

APROS5 model of PACTEL was created as a Master's Thesis work using the new graphical user interface GRADES. The model was basically created from scratch using only the plant design data. The nodalization of the simulation model was defined by the modeller. The aim of such an approach was, in addition of creating the new model, to test the new user interface and to test APROS modelling capabilities when used by a novice in thermal hydraulics. Another aim of such a procedure was to avoid the possible errors embedded in the previous models. The current paper describes the model created and preliminary validation calculations made with the model. The paper contains also a short overview on the PACTEL models created with APROS and the validation calculations performed.

APROS validation using PACTEL results

Several models of the PACTEL facility have been created using APROS (Ahvonen 1993, Plit 1996, Leppänen 2000). The PACTEL facility has been modelled also using several RELAP5 versions and these models have been used as source of information in various amounts when building the APROS models. The first APROS model (Ahvonen 1993) was created using the five-equation thermal hydraulics. The second model (Plit 1996) was created using the six-equation thermal hydraulics. The current, and most extensive model (Leppänen 2000), was also created using the six-equation thermal hydraulics. The basic models have been modified by other users, in order to calculate new experiments. Altogether 18 PACTEL experiments presented in table 6 have been calculated with APROS. Some of the experiments have actually been calculated with several successive models, since these experiments, like LOF-10 and SBL-22, are suitable for judging the basic functionality of the model.

Table 6. PACTEL tests calculated with APROS code.

Test type	**Test identification**
• Natural circulation experiment	⬧ ISP-33
• Small Break LOCA experiment, hot leg loop seal behavior	⬧ SBL-22, SBL-30
• Small Break LOCA experiment, validation of the accumulator models	⬧ SBL-31
• SBLOCA experiment, validation of the accumulator and HPI models	⬧ SBL-33
• Single phase flow instabilities	⬧ CMP-04, CMP-08, CMP-09
• Horizontal steam generator behavior	⬧ SG-2, SG-3, SG-4
• Loss of feedwater, horizontal steam generator behavior	⬧ LOF-10
• Gravity Driven Core Cooling test (GDE-11)	⬧ GDE-11, GDE-24, GDE-34, GDE-41, GDE-43
• ATWS-test (Compressibility of steam in the top of pressuriser, pressuriser heating)	⬧ ATWS-10

APROS5 model of PACTEL facility

The model was created with the GRADES user interface using the test facility description (Tuunanen et al. 1998). The entire model consists of 15 GRADES nets that are connected to each other. Each net contains a part of the PACTEL facility that consist of core, downcomer and lower plenum, three loops, steam generator, pressurizer, emergency core cooling system and heat losses. The nodalization was principally defined by the modeller.

The model was built using APROS components like pipes, points, nodes etc. The core was described with design reactor and design reactor heat structure components. The steam generators and the pressurizer of PACTEL were modeled with the process component nodes, since the built-in steam generator and pressurizer process components of APROS were deemed to be too robust for this purpose. Automation components were used to model the control systems in the test facility.

The core-net includes reactor core and the upper plenum. This net has also the necessary modules to control the APROS simulation session and the modules to control and define the output files of the simulation results. The core-net is presented in the figure 66.

The core model includes heating elements and thermal hydraulic channels. The model includes also the description of the heat flow from the different core segments into the bypass channel, to the pressure vessel and finally to the air outside. There are also some heat structures to model the heat losses from the upper plenum and from the junction between the core and the lower plenum. The core-net also includes the upper plenum consisting of the volume above the core, the hot leg connections and the upper plenum diffusor. The upper plenum is mainly modelled with a heat pipe component. The upper plenum diffusor is modelled with node components and pipes. The purpose of the diffusor is to prevent the high pressure injection system from injecting the water straight into the hot legs.

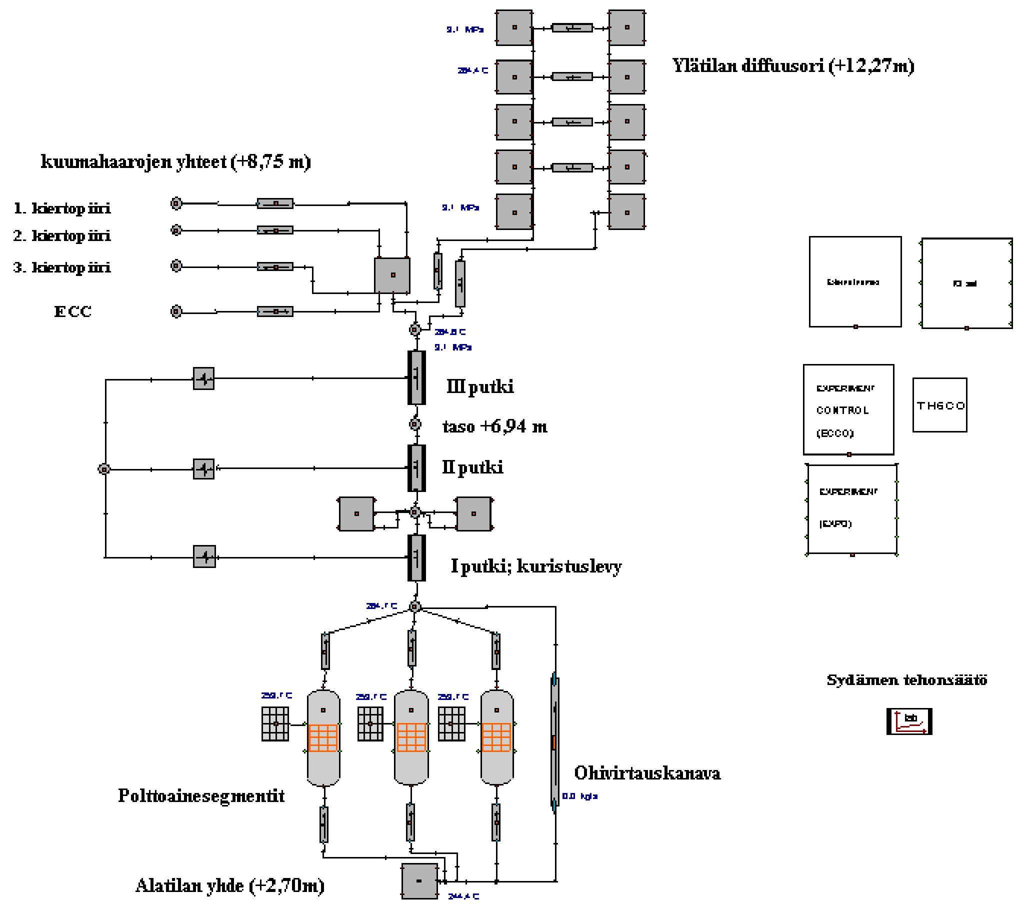

Figure 66. Core-net.

In the PACTEL facility the downcomer and lower plenum are modelled with an u-tube structure where the downcomer and the lower plenum are connected together. In the APROS model these have been described with heat pipe components that have different diameters.

The hot and cold leg of one loop are modelled in the same GRADES net. Thus each one of the three loop-nets composes a whole loop without a steam generator. The loops are constructed with heat pipe components. Heat losses from the loops can be described also with these components. The loops include shut-off valves and pumps. There is also a main circulation pump in each cold leg modelled with a basic pump component. The pump is controlled with a device control component which controls the rotation speed of the pump. The correct mass flow is set by adjusting the rotation speed.

The PACTEL steam generator model consists of two parts, the primary side and the secondary side. Both sides were divided in five vertical layers in order to take into account the water level variations. The primary side includes hot and cold collectors and heat pipes. The collectors, as well as the secondary side were build with node components of APROS, since these allow a very detailed modelling of the steam generator. Different nodes are connected to each other with pipe connections. In the model, there is only one steam valve controlling the whole secondary side pressure. All three steam generators are connected to one common pipe line where steam valve is located. By this way, all three steam generators can be kept exactly under the same pressure. Also, the simulation time is reduced as a result of fewer nodes and control systems. Figure 67 shows an example of the steam generator net.

The pressurizer of the PACTEL facility was also modelled using node components in order to achieve a more detailed description than with the build-in pressurizer process component. Heaters in the pressurizer were modelled with heat structure components of APROS. The pressurizer pressure control system and spray system were also included in the model using APROS control modules and binary switches.The model included also the emergency core cooling systems (ECCS) of the PACTEL facility. These were not used in the two validation cases.

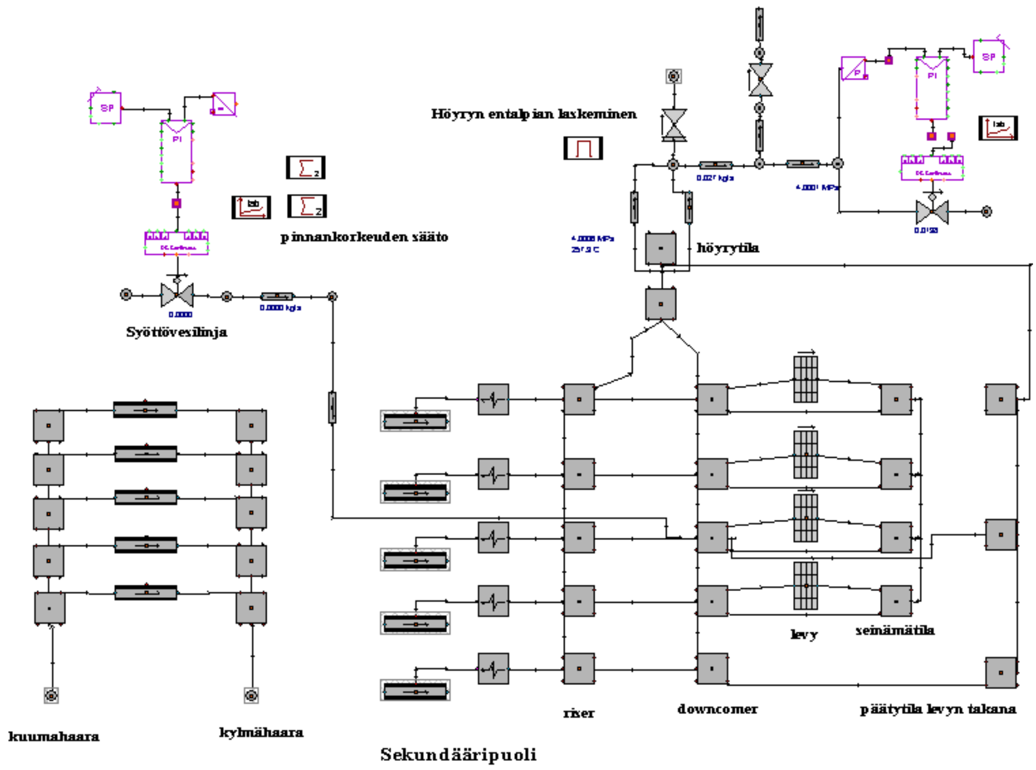

Figure 67. Steam generator -net.

In the construction of the new APROS5 model of PACTEL considerable effort was put on modelling all the heat losses of the experimental facility. The heat losses were added to the simulation model by attaching heat structures to the proper volumes and by conducting the heat through them to a point which represents the air outside the test facility. The heat losses could only be set according to the best estimate values and by using modellers own judgement and intuition (Leppänen 2000). In the facility the heat losses originate from several sources and they are distributed around the model. The correct value for heat losses is extremely important in test facility, where surface are ratio to water volume is higher than in real power plant. Heat losses, especially in the pressurizer, have an effect on pressure behavior in long tests, like the ones shousen for this study. The heat losses were modelled in two nets. Figure 68 shows the system level heat losses. Singular heat losses were included in another net.

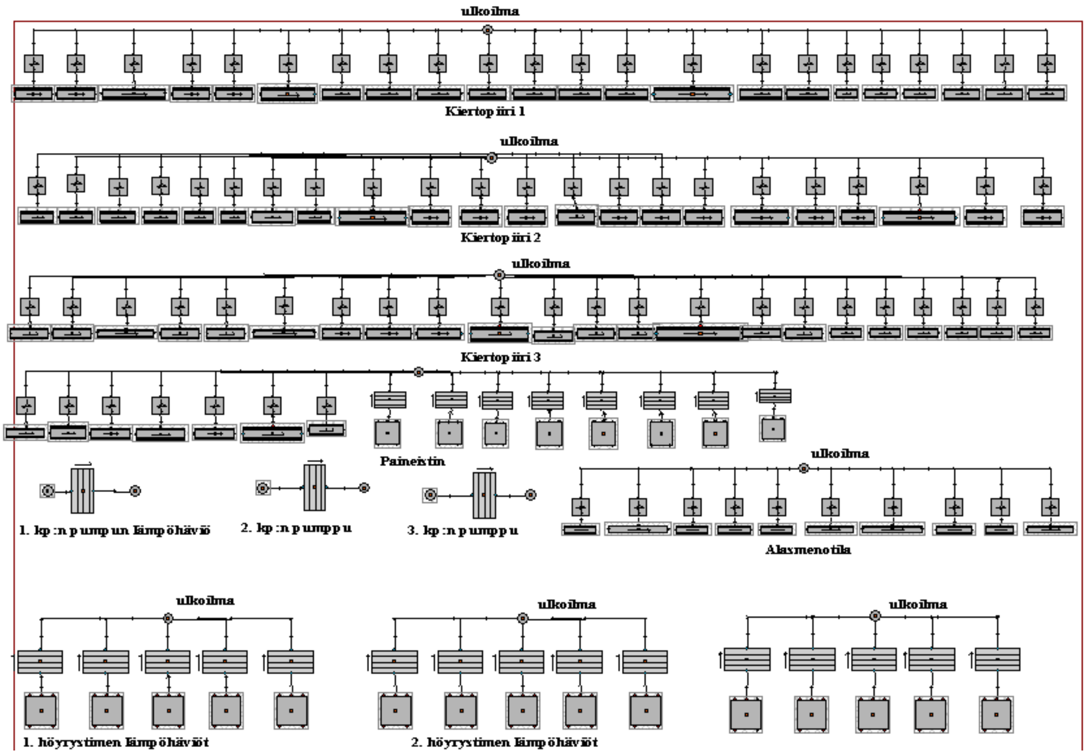

Figure 68. Heat loss1 -net.

Test calculations

Validation calculations included Loss of Feedwater test LOF-10 (Kouhia & Puustinen 1998) and calculation of Small Break Loca experiment SBL-22 (Riikonen et al. 1994). The LOF-10 test was selected as the first validation case, since the test itself is rather simple and easy to calculate, but reveals the possible errors and discrepancies in the model. The essential point in these comparisons was the qualitatively correct behaviour of the model with description of all physical phenomena involved. The results obtained have been described and discussed in detail in (Leppänen 2000). The results indicated clear needs for improvements in the model. These include the heat losses in pressurizer and in primary circuit and flow loss coefficients. Tuning of the flow loss coefficients and heat losses would require more detailed heat and pressure loss measurements. It was observed that by improving the noding of the tube bundles improvement in the calculation results was obtained. APROS was able to calculate all phenomena involved, and the physical phenomena behaved in the expected manner without oscillations.

The second validation case was the Small Break LOCA experiment SBL-22. In the experiment the behaviour of the steam generator was studied with various amounts of primary coolant. The experiment was performed using one loop and one steam generator. The experiment was started with a 1000 seconds steady state period. Then the main circulation pump was stopped and the leak was opened. Simultaneously the pressurizer was isolated from the loops by closing the pressurizer shut-off valve. After the stopping of the main circulation pump natural flow conditions were established. The leak flow decreased the total inventory of primary coolant, and thus the flow changed into two-phase flow and thereafter into steam flow. The primary pressure was changing during the experiment, but the secondary pressure was kept constant. The leak size was 1 mm or 0.04% of the cross section area of PACTEL cold leg. The leak size corresponds to 0.12% of the scaled flow cross section of the reference plant.

Calculated and experimental results for the upper plenum pressure, core coolant inlet temperature, cold leg temperature, core coolant outlet temperature, hot leg temperature, loop mass flow, leak mass flow and integrated mass flow have been compared in (Leppänen 2000). The upper plenum pressure has been presented in figure 69.

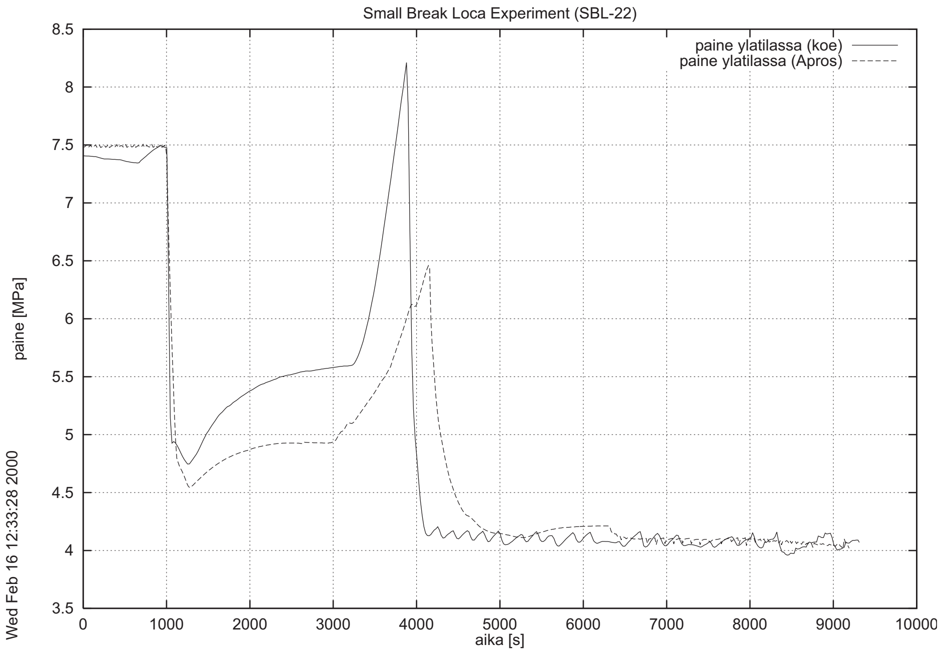

Figure 69. Upper plenum pressure.

It was observed that the calculated upper plenum pressure behaviour agreed with the experimental result qualitatively, but the pressure peak was somewhat delayed and the height of the pressure peak was lower than in the experiment. Similar behaviour was indicated in the core coolant outlet temperature that has been presented in figure 70.

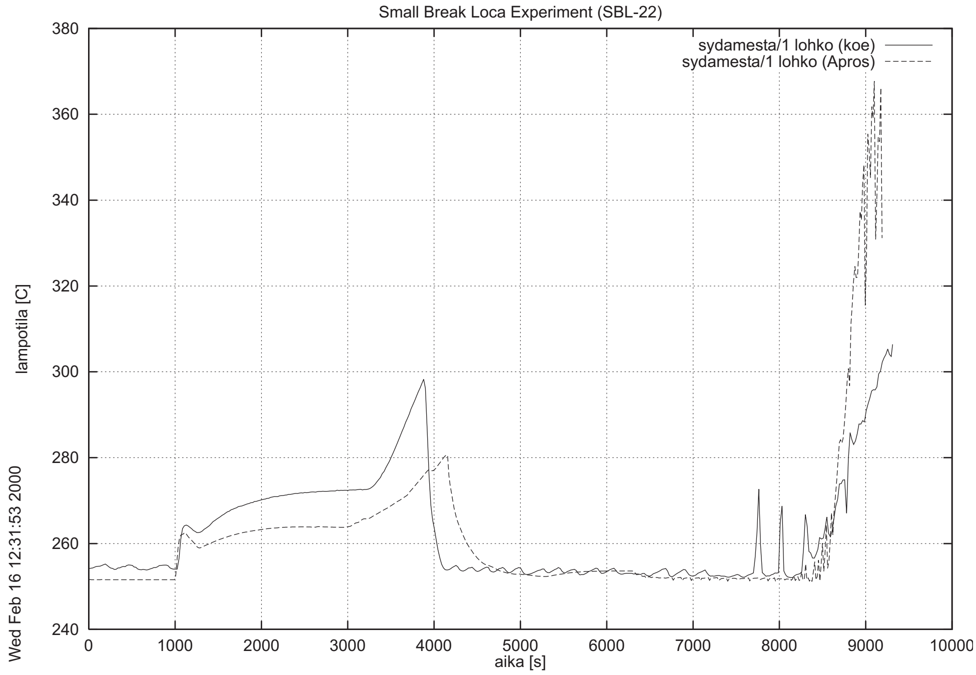

Figure 70. Core coolant outlet temperature.

The same trend of delayed and somewhat damped behaviour of the calculated parameters in comparison with the measured values was observed also in the other parameters. According to (Leppänen 2000) the probable principal reasons for the differences between the calculated and measured values were in the ratio of the flow resistances between the bypass channel and core, and in the modelling of the heat losses. In APROS the flow resistances are described with loss coefficients of flow. Leppänen (2000), acticipated that the loss coefficient of the bypass channel was too low. Thus, too much coolant was by-passing the core without heating, and consequently the coolant temperature in primary circuit remained too low. Due to the too low coolant temperature the pressure peak was delayed and the peak height was lower than in the experiment. Another parameter having a crucial effect on the results were the heat losses. It was found out that an error in the heat loss of one single component could have a very significant effect on the results obtained even in cases where the total heat loss was correctly modelled. The third parameter affecting the calculated results was found out to be the leak mass flow. The difference of the calculated and actual leak mass flow has a decisive effect on the results obtained in particular at the end phase of the experiment. The model predicted a higher circulation mass flow than actually measured. A similar feature was observed also in the calculation of the LOF-10 experiment.

Discussion

Case SBL-22 turned out to be much more difficult to calculate than the case LOF-10. In calculation of case SBL-22 clear differences between the experimental and calculted results remained. Greatest problem was found to be tuning of the leak mass flow rate. Calculation of case SBL-22 indicated also that the circulation flow between the heated core parts and by-pass channel was not correctly modelled. According to (Leppänen 2000) improved results could be obtained by tuning the loss coefficients of the core and by-pass channel. However, this was considered to be beyond the original purpuse of the Master's Thesis work.

The experiments LOF-10 and SBL-22 have been used also in testing the previous APROS models of PACTEL facility. In a previous work (Plit 1996) very good agreement was found between the calculated and measured results for the case SBL-22. In that work the testing of the model was much more extensive, and the model produced good results also in blind test calculations of PACTEL tests SBL-30 and SBL-31. Both the current model and the model previously created (Plit 1996) describe the PACTEL facility using six-equation thermal hydraulics. The current model is more detailed than the previous APROS models of PACTEL. The current model has 454 thermal hydraulic nodes, 513 branches and 1478 heat structure nodes. Reference (Plit 1996) does not indicate directly the model size, but references (Vihavainen 1998) where this model was used indicates model size of 390 thermal hydraulic nodes and 1347 heat structure nodes. Thus, the model size has increased only with some 20%. Tuning of an extensive model is more laborous than tuning of a simple model as long a some key parameters are considered. However, only a detailed model can describe well all the phenomena observed in the experiments. Another difference between the current model and the model created previously (Plit 1996) was that the current model was originally intended to be created totally by the modeller using only the facility design data. The creation of the previous model was based on a documentation of the PACTEL facility made by two thermal hydraulic experts for the participants of an international standard problem (Purhonen & Miettinen 1992). It can be scpeculated that use of such a document could guide to a more successfull noding than when a student is using the original plant data and drawings.

The graphical user interface should have no effect on the results obtained. However, during the creation of the APROS5 model of PACTEL the GRADES was still at its prototype phase causing immensense practical problems for the modeller, and increasing the time required for the model creation. The creation of the previous model took place with the GRINAP user interface that was already a proven standard tool.

Summary

The paper presents a short summary of the APROS5 model of PACTEL test facility that was created in a Master's Thesis work (Leppänen 2000) and the preliminary validation calculations. The standpoint of the work was clearly to test both the new graphical user interface GRADES and the capability of APROS in the hands of a novice in thermal hydraulics. The result was a detailed and working model of the PACTEL facility that after some additional tuning is ready for further calculations also in context of planning new experiments.

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5.9 MOSES special report

Hydrogen behaviour in Olkiluoto BWR reactor building during a severe accident

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Abstract

Hydrogen behaviour in Olkiluoto reactor building during a severe accident has been assessed. The objective of the work was to investigate, if hydrogen can form flammable and detonable mixtures in the reactor building, evaluate the possibility of onset of detonation, assess the pressure loads caused by detonation, and study the integrity of walls and pipe penetrations in the reactor building.

The initial accident sequence was station blackout with depressurisation of the reactor coolant system. A conservative assumption of 100% zirconium oxidation in a core was made. The hydrogen was assumed to leak from the containment into the reactor building from the pipe penetration in the upper part of the containment. Three different rooms in the reactor building and two different leakage sizes were considered.

The analyses were made with the MELCOR (initial conditions), FLUENT (hydrogen accumulation, deflagration, flame acceleration), DETO and DET3D (detonation loads) and ABAQUS (structural analysis) codes. The analyses indicated strong accumulation of hydrogen to the upper parts of the reactor building rooms under consideration. Combustible conditions existed in all analysed cases somewhere in the room. The simulations and semi-empirical analyses indicated that the criteria for the onset of detonation are reached in some, but not in all cases considered. The possibility of deflagration-to-detonation transition (DDT) in the reactor building could not, therefore, be excluded. First results of three-dimensional detonation studies for reactor building room B.60.80 showed that the highest pressure spikes of about 10 MPa occurred in the corners of the room. Corresponding pressure impulses were about 30–35 kPa-s. Structural analyses showed that the concrete wall of the room may survive the detonation peak transient, but the relatively slowly decreasing static-type pressure after the peak detonation damaged the wall much more severely than the short detonation peak pressure. Further work consists of the analyses of integrity of the containment pipe penetration in the room B.60.80 under detonation loads.

Introduction

The containments of Olkiluoto BWR units (figure 71) are inerted with nitrogen during normal operation, and hence, hydrogen burning has not been considered a major problem in severe accidents. The BWR core contains a large amount of zirconium. Significant release of hydrogen to relatively small containment can occur in a severe accident. Taking into account the thermal and pressure loads, hydrogen leakage from containment into the surrounding reactor-building rooms can not be excluded. The atmosphere in the reactor building is normal air, therefore, ignition and combustion of hydrogen is possible. The safety concern is, if the hydrogen in the reactor building can detonate and jeopardise the containment integrity from outside.

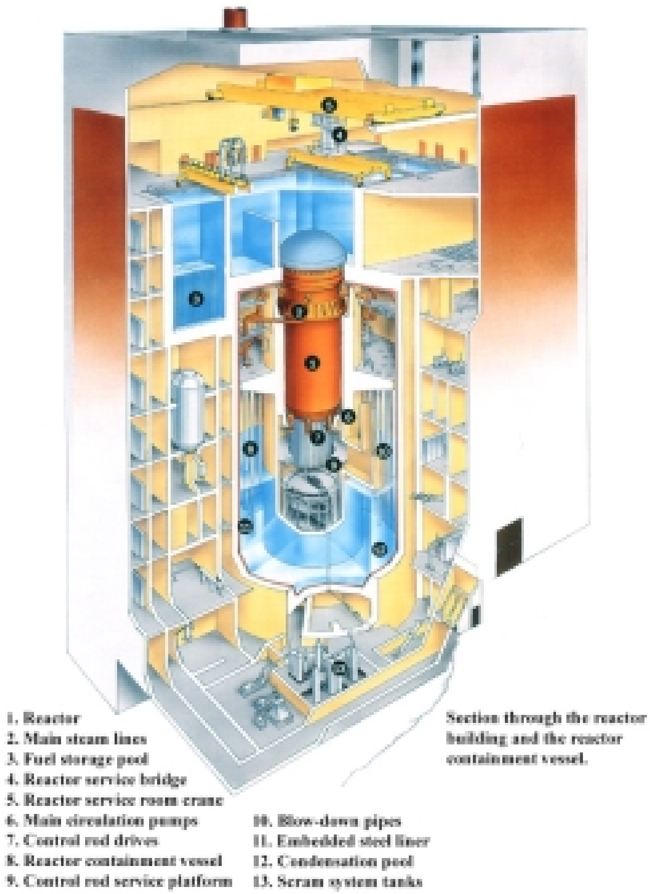

Figure 71. Olkiluoto BWR containment and adjacent reactor building rooms (Manninen et al., 2000).

Technical Approach

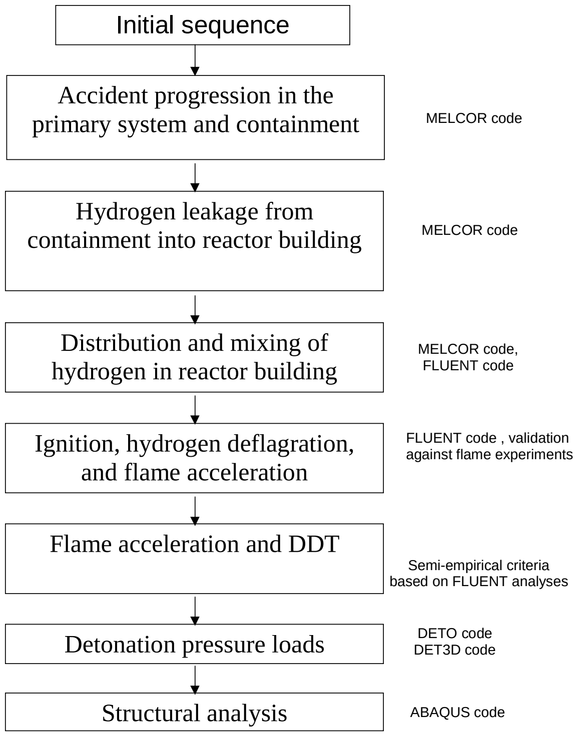

Figure 72. Stages of research of hydrogen behaviour in Olkiluoto reactor building.

The hydrogen behaviour in Olkiluoto reactor building was extensively studied step by step using the procedure shown in figure 72. The initial accident sequence was assumed to be a station blackout accident with depressurisation of the reactor coolant system. The progression of accident and the hydrogen leakage rate from the containment into the reactor building were calculated with the MELCOR 1.8.4 code (Sandia National Laboratories 1994). 100% of zirconium was conservatively assumed to oxidise. This results in total hydrogen release of 1900 kg. MELCOR analyses were also used in the first estimates for hydrogen distribution in the reactor building. They also provided the initial conditions for further, more detailed hydrogen accumulation studies. These studies were carried out with the three-dimensional CFD code FLUENT (Fluent Inc. 1998). The FLUENT code was also used to investigate hydrogen deflagration and flame acceleration in reactor building room B.60.80. The possibility of onset of detonation was estimated applying simple semi-empirical rules for flame acceleration and deflagration-to-detonation transition (DDT) in conditions based on previous FLUENT analyses. The first estimates for the pressure loads under hydrogen detonations were carried out with a simple computerised hand-calculation type method code DETO. More detailed, best-estimate analyses of hydrogen detonations were then performed with three-dimensional finite difference code DET3D. Finally, the ABAQUS code (ABAQUS 1998) was applied for structural analyses of the reactor building (Saarenheimo 2000).

Hydrogen leakage from containment and accumulation in reactor building

The first scoping analyses of hydrogen leakage from the containment and distribution in the reactor building were carried out with the MELCOR code (Manninen et al. 2000). Significant zirconium oxidation and hydrogen generation lead to pressure buildup in the containment and leakage to the surrounding reactor building. MELCOR is based on the lumped parameter approach based on averaging of different processes within a control volume. Even the assumption of homogeneous mixing inside a control volume resulted in flammable hydrogen concentrations in some reactor building rooms. These results indicated that further, more detailed CFD analyses were therefore needed.

Three-dimensional CFD-simulations of the hydrogen accumulation in two different reactor building rooms were performed with the FLUENT 5.0 code (Manninen & Huhtanen 1999). The rooms are large (2164 m^3 and 897 m^3) and tall having complicated geometries (figure 73). Two different leak sizes were considered, 2 mm^2, which corresponds to the nominal leakage of containment, and a large leak of 20 mm^2. After about 13 000 s from the beginning of the hydrogen release, the total mass of hydrogen leak to the reactor building was about 3 kg and 30 kg, respectively. The leak was assumed to occur through a single hole to the same reactor building room. This assumption is conservative, because in reality the leakage would rather be a diffuse one involving several rooms. The leak was assumed in the upper part of the rooms where the pipe penetrations exist (figure 74).

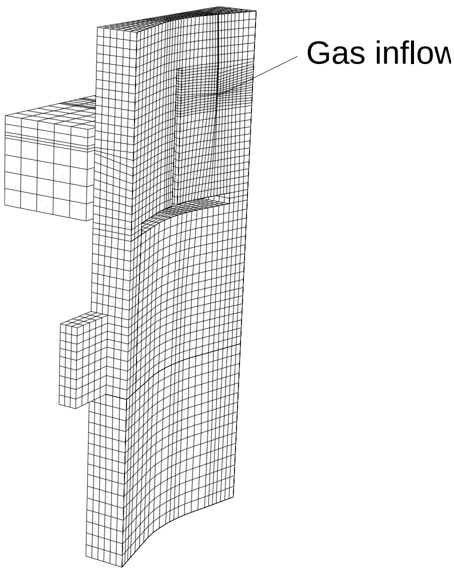

Figure 73. Computational grid for the room B.60.80 (Manninen et al. 2000).

Figure 74. Pipe penetration in the containment wall in rector building room B.60.80 (Manninen et al. 2000.)

The flow in FLUENT simulations was expected to be dominated by buoyancy due to density variations. This is justified because the density of the incoming gas was only about 20% of the initial gas density in the reactor building room under consideration. The solution techniques consist of the second order accurate upwind discretisation for the convection terms in the momentum, energy, turbulence, and species equations. The second order implicit discretisation for the time integration was used. Pressure values at the cell faces were interpolated using a body-force-weighted interpolation scheme (Fluent inc., 1998). The pressure-velocity coupling was solved using the SIMPLEC algorithm. Turbulence was modelled with the standard k-ε model with buoyancy terms included.

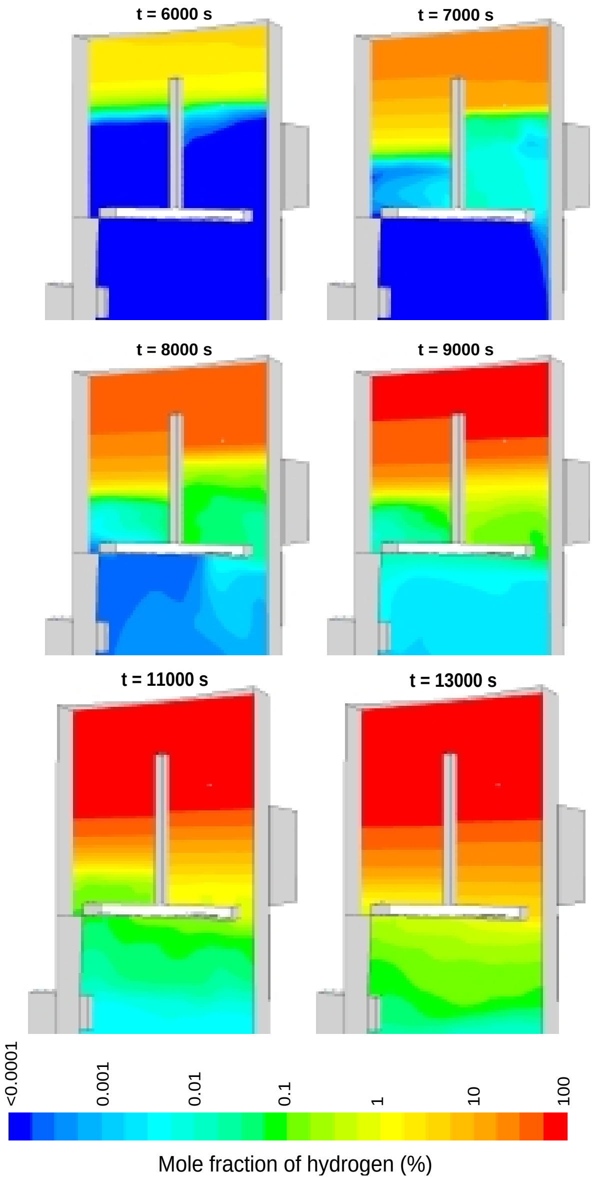

Figure 75. Mole fraction of hydrogen in the room B.60.80 at various times. 20 mm^2 leakage (Manninen et al., 2000).

The hydrogen accumulation analyses with FLUENT showed that hydrogen collected mostly to the volume above the gas inlet leading to stabile hydrogen stratification and accumulation close to the ceiling (figure 75). In case of 20 mm^2 leakage, the mole fraction of hydrogen reached the maximum value of about 80% in the upper parts of the room B.60.80. At the same location, the mole fraction of oxygen was less than 1%. This kind of strong stratification is important from the point of view of the possible combustion of hydrogen. Depending on the size of the leak and the timing of hydrogen ignition, several possible combustion scenarios could be possible. The larger the leak was, the sooner the mixture became flammable, and at a later stage also detonable.

Hydrogen deflagration and flame acceleration

The hydrogen accumulation analyses suggested that hydrogen deflagration and flame acceleration should be investigated. This was carried out for one selected reactor building room B.60.80 using the FLUENT code (Manninen et al. 1999). The main objective of these analyses was to assess whether the flame can accelerate in the reactor building. This information was necessary in order to assess the possibility of onset of detonation in the reactor building.

The FLUENT parameters for deflagration and flame acceleration studies were first validated against the FLAME experiments (Sherman et al 1989). The facility consists of a 30.5 m long rectangular channel closed on the ignition end and open on the far end. The calculations were carried out with three different grids by varying the time step. The cases both with and without obstacles were considered (figures 76 and 77).

In general, the simulation of the FLAME experiments showed good qualitative agreement with the measured results.

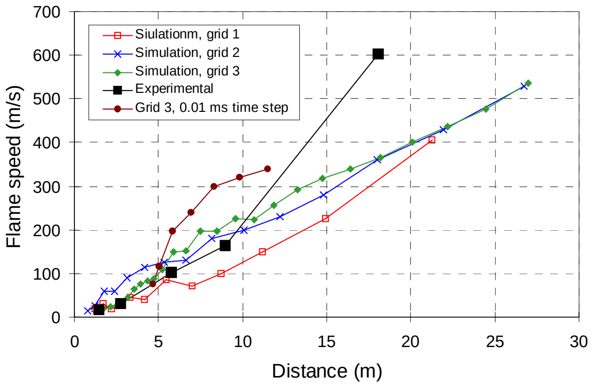

Figure 76. Simulated and measured flame speed in FLAME tst 22 (with obstacles) (Manninen et al. 1999b).

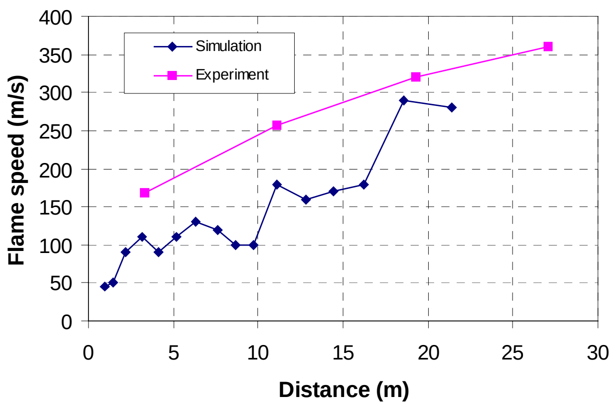

Figure 77. Simulated and measured flame speed in FLAME test 8 (without obstacles) (Manninen et al. 1999b).

Independence of the computational grid or applied time step were not achieved. In any case, the results were qualitatively sufficient accurate to justify the use of the simple FLUENT modelling for the first scoping study of the hydrogen deflagration and flame acceleration in the Olkiluoto reactor building.

Three different hydrogen distribution situations in room B.60.80 were selected from previous FLUENT calculations (Manninen & Huhtanen 1999) for the deflagration and detonation assessment. Case 1 was the end state of 20 mm^2 leakage where the hydrogen mole fraction close to the ceiling was very high (about 80%) and oxygen fraction was low (< 1%). Case 2 corresponded to the earlier instant of time of 20 mm^2 leakage when the average mole fractions of gases in the volume above the gas inlet were roughly stoichiometric. Case 3 corresponded to 2 mm^2 leakage where the hydrogen and oxygen mole fractions were both roughly 17% near the room ceiling.

The combustion rate in the deflagration analyses was assumed to be limited by the turbulent mixing. This assumption relied on the estimate that the kinetic reaction rate is larger than the rate due to turbulent mixing and can be omitted in a qualitative study. The turbulent mixing was computed using the eddy-break-up (EBU) model.

The simulation of hydrogen deflagration and flame acceleration in Olkiluoto reactor building room B.60.80 was carried out utilising the experience obtained from FLAME experiments. The simulation suggested that the flame front maximally accelerates to a limiting value on the order of 500–600 m/s (figure 78). However, definite conclusion on flame acceleration could not be drawn from the FLUENT simulation because of the simplification and limitations in the modelling. The validation against the FLAME experiments indicated that FLUENT may underestimated the flame speed, especially in the high-speed stage.

Location of ABAQUS
FEM model

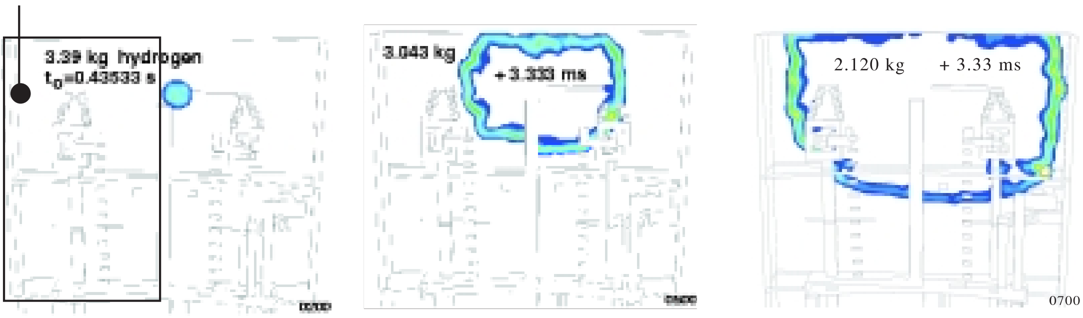

Figure 78. Propagation of the flame front in the upper part of the room B.60.80. Contours of the reaction rate are shown in range 0.1 kg/m^{3}s (blue) – 10 kg/m^{3}s (red) (Manninen et al., 1999).

Onset of detonation

Direct initiation of detonation by an energetic source is not believed to be very probable in NPP containments and reactor buildings due to very high critical energy needed (Guirao et al. 1989). More probable initiation mechanism is therefore an indirect initiation e.g. deflagratio-to-detonation transition (DDT). The possibility of DDT in Olkiluoto reactor building was assessed by applying simple, conservative semi-empirical criteria for flame acceleration, for deflagration-to-detonation transition (DDT) and for propagation of detonation proposed by Breitung et. al. (1999). The criteria can be considered as necessary, but not sufficient conditions for the flame acceleration and DDT.

The criteria were applied by utilising the results from previous deflagration and flame acceleration analyses with FLUENT code (Manninen et al. 1999). The first condition necessary for DDT states that the expansion ratio (density of unburned gases divided by the density of burned gases) must exceed the value 3.5–4. In both simulation cases under consideration, the expansion ratio attained values 3–4, thus the possibility of detonation could not be ruled out. According to the second condition, the flame has to accelerate to a flame Mach number (flame speed normalised by the sound speed in the unburned gases) of 1.5. In Case 1 the simulation predicted 1.4–1.6 for the flame Mach number. In case 2 the Mach number remained in a lower value of 1–1.2.

In Case 1, the simulations and semi-empirical analyses indicated that the necessary criteria for the onset of detonation are just reached. However, part of the hydrogen is already burned when the flame has accelerated to the critical value. In particular, the upwards propagating flame is close to the upper limit of the detonable region. In Case 2, the flame speed seemed to stay below the critical value, although the gas composition was close to the optimal for combustion. The likely explanation for the result is the fairly open geometry in the upper part of the room B.60.80. Turbulence needed to accelerate the flame is not produced. The onset of detonation from flame acceleration seemed therefore unlikely. CFD-simulation of Case 3 was not carried out, but it is expected to behave qualitatively in the same way as Case 2.

The FLUENT simulations and semi-empirical analyses indicated that the criteria for flame acceleration and DDT were just reached in reactor building, but not in all cases under consideration. The conclusion was that the possibility of detonation in the reactor building could not be ruled out, and further analyses were justified in order to assess the pressure loads, if hydrogen detonations occur.

Detonation pressure loads

The first rough approximation for the shock pressure loads caused by detonation was made assuming a direct initiation of detonation and applying the strong explosion theory (Silde & Lindholm 2000). The simple 1-D model (DETO code) gave the first estimate of the effect of incident shock and the first reflection of shock wave on wall structures. Comparing the results to the theoretical Chapman-Jouguet detonation values indicated that the simple approach partly gave very conservative values and overestimated the peak pressures. Due to the limitations of the approach, and in order to take into account the multiple shock reflections and focusing of shock waves, more detailed three-dimensional detonation analyses were found to be necessary.

Three-dimensional detonation analyses were carried out with the DET3D code developed at FzK (Breitung & Redlinger 1993). The code uses the finite difference method based on Euler equations for a multicomponent reacting gas. Detonations in Olkiluoto reactor building room B.60.80 were calculated in three different initial conditions obtained from previous FLUENT analyses of hydrogen accumulation. Also the location of ignition was varied.

The detonation simulation indicated that a spherical shock wave propagating with velocity close to the Chapman-Jouguet value was formed after ignition (figure 79) (Silde & Redlinger 2000). Highest pressure spikes existed in corners of the room. The first analyses of the results show that the simulated detonation pressure of incident shock is about 1.5 MPa in stoichiometric gas mixture, while the maximum pressure spikes in the corners are about 7–10 MPa depending on the case considered (figure 80). The pressure maximum is affected by multiple reflections and focusing of shock waves near the corners. Maximum pressure impulses to structures during a 150 ms simulation are about 30–35 kPa-s (figure 81). Further analyses of the detonation calculations are currently under way.

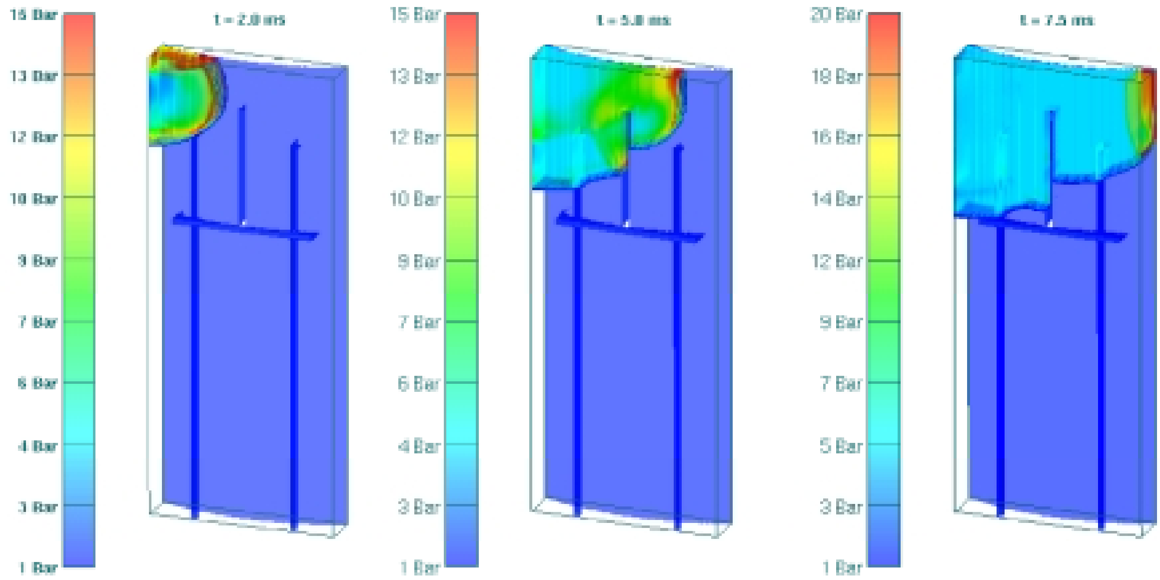

Figure 79. Countours of pressure in room B.60.80 at three instants of time during detonation, 2.0 ms (left), 5.0 ms (middle), and 7.5 ms (right).

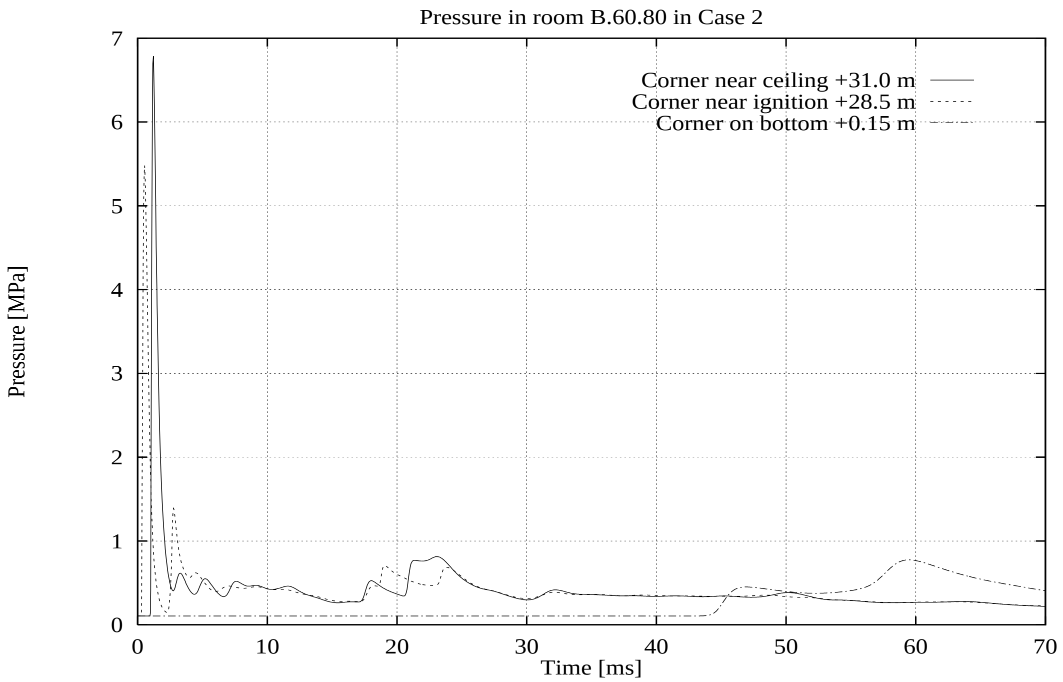

Figure 80. Calculated pressures at three different elevations in room B.60.80. Case 2.

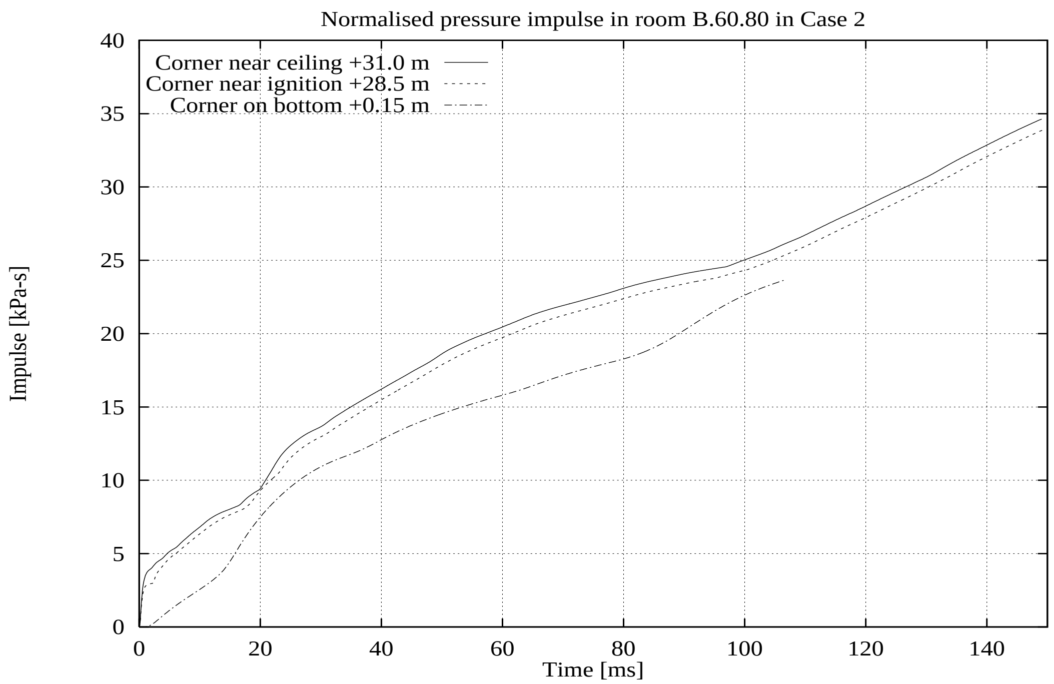

Figure 81. Calculated pressure impulses at three different elevations in room B.60.80 during 150 ms simulation. Case 2.

Structural integrity

The structural analyses were carried out with the ABAQUS finite element (FE) code using as input data the pressure loads obtained from detonation analyses with a simple DETO code. The structural integrity of the wall was studied under detonations corresponding the detonable hydrogen mass of 0.5 kg and 1.428 kg. The ignition was assumed to occur in the middle gas space in the upper part of the room B.60.80. As seen in figure 80, hydrogen detonations lead to high and short, peak type pressure transients, which are followed by a relatively high and slowly decreasing pressure (Silde & Lindholm 2000) and (Silde 1999).

Linear analyses were carried out by ABAQUS/Standard code and the load carrying capacity was evaluated based on codes and standards. Only a part of the reinforced concrete wall was considered in non-linear dynamic FE analyses carried out using ABAQUS/Explicit code. The location of this FE model is indicated in figure 78. Simplified boundary conditions were used for simulating the effect of surrounding structures. The upper edge of the FE model was assumed to be free and there was a symmetry line assumption along the right edge of the model. The lower edge and the left edge were modelled as fully fixed. The reinforcement and its rate dependent elastic-plastic material behaviour as well as the tensile cracking of concrete were modelled.

Crack development at the outer surface of the wall, due to the energy release from burn of 1.428 kg hydrogen is presented in figure 82. The cracking of the inner surface is shown in figure 83, respectively. The ignition was assumed to occur at the symmetry line, near the upper edge of the wall. After extensive cracking during the detonation peak, the energy balance was lost and further results were not more reliable.

The wall resisted quite well the pressure increase before the detonation. This pressure value was near the 'design load'. The duration of the detonation was short compared with the eigenperiods of the wall. The wall may survive somehow a shock pressure load corresponding the energy release from burn of a relatively small amount of hydrogen. The slowly decreasing static type pressure after the detonation peak damages the wall much more severely than the detonation peak (Saarenheimo 2000).

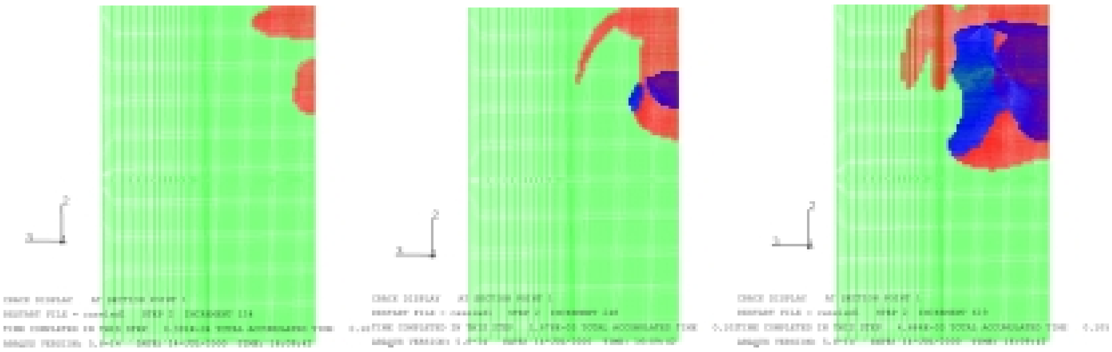

Figure 82. Cracks at the outer surface of the wall, t=0.101 s, t=0.102 s and t=0.105 s.

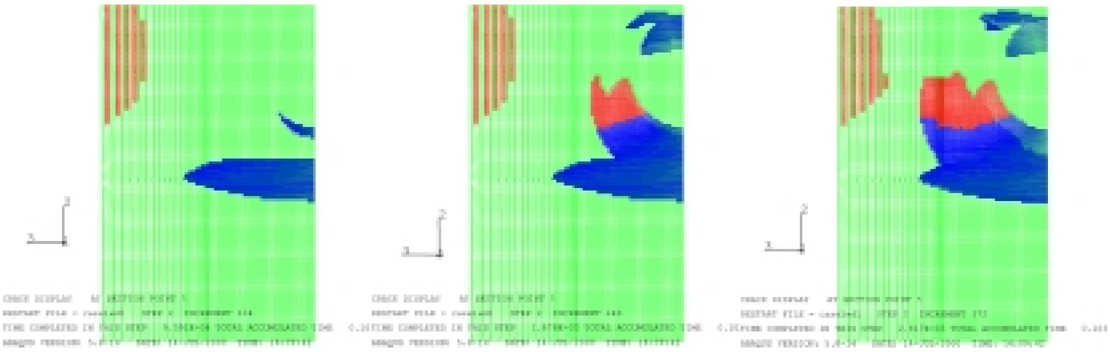

Figure 83. Cracks at the inner surface of the wall, t=0.101 s, t=0.102 s and t=0.105 s.

Conclusions and future work

The hydrogen behaviour in Olkiluoto reactor building has been extensively studied. Significant zirconium oxidation and generation of hydrogen during a severe accident leads to pressure buildup in the containment. Hydrogen leakage to the surrounding reactor building may occur. If the ventilation of reactor building is out of operation, as is the case in station blackout accident, hydrogen tends to accumulate close to the ceilings in the reactor building rooms. Depending on the location and size of leakage, the hydrogen accumulation may form flammable and even detonable mixtures. Definite conclusion on flame acceleration and DDT could not be drawn because of the simplification and limitations in the modelling. The assessment described in this work could not rule out the possibility of flame acceleration and DDT in station blackout accident. If the detonation ignited, the highest pressure loads were simulated to occur in the corners of the room in the cases under consideration. The concrete wall of the reactor building room B.60.80 may survive somehow the detonation peak pressure, but the relatively slowly decreasing static-type pressure after the peak detonation would damage the wall much more severely than the detonation peak.

The first structural analyses were based on the shock pressure loads calculated with a simple DETO code. The future work consists of the assessment of integrity of containment pipe penetrations in reactor building room B.60.80 using more realistic detonation loads obtained from the DET3D finite difference code.

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5.10 FISRE special report

Insulation resistance of electronic circuits in smoke – quantification of the problem and needed countermeasures

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Abstract

Acute effects of smoke on insulation resistance of electronics used in programmable automation circuitry in nuclear power plants were studied experimentally. Real and mock-up circuitry was exposed to smoke in four tests. Relevant physical parameters of smoke and electrical performance of circuitry was measured online for a period of an hour. It was noticed that insulation resistance decreased by three orders of magnitude

due to soot deposits on surfaces on uncoated circuitry. For real commercial circuits, coated by a protective lacquer layer, no electrical changes were observed.

A quantitative model for smoke exposure and deposition on surfaces is proposed based on existing physical models of aerosols. A consistent picture of the relevant processes was found, and an easy calculation method for smoke hazard estimation for PSA-work is proposed for insulation resistance deterioration. This model allows also estimation of the required performance of protective coatings to prevent parasitic leakage currents between different parts of circuitry.

Introduction

The experiments were designed around a standard smoke exposure chamber (ISO 5659-2:1994). Some auxiliary features were added to meet some of the requirements expected from earlier work at VTT and Sandia National Laboratory. Only acute effects of smoke were considered and slower processes like corrosion are not included.

For characterising changes on electronic circuits due to soot microscopic properties of soot are needed. Smoke is understood as the visible part of fire effluent. It is an aerosol, where small particles of solid or liquid are suspended in air. The main element of the particles (soot) is carbon. Single smoke particles are usually conglomerates of smaller spherules about 30 nm in diameter forming complicated space structures (Mulholland 1995). The effective diameter extends from tens of nanometers to tens of micrometers, sometimes even higher. A typical value of the geometric mean volume diameter falls in the interval 0.3–3 μm. Soot smoke deposits on surfaces form an electrically conducting film, which could deteriorate sensitive electronics.

Particle sizes from PVC smoke have been determined under different conditions (Bankston et al. 1977). The size distribution is roughly logarithmically normal (Mulholland 1995) with a mean of 0.6 μm for smouldering PVC. For flaming combustion particle size distribution has a maximum at 0.2 μm.

The size distribution is not constant but varies somewhat with time. This temporal behaviour is understood in terms of smoke particle coagulation and settling. Settling removes larger particles more efficiently than small particles. Coagulation is a phenomenon where two colliding particles stick together after collision.

In our experiments polychromatic light extinction was measured. The intensity I_0 of the pencil of a light ray attenuates after transmitting through a pathlength L of the smoke to intensity I as

$$\frac{I}{I_o} = \exp(-kL) \quad (32)$$

The mass concentration of smoke c_m is obtained from direct optical measurements as

$$c_m = \frac{1}{\sigma_{ext}} \frac{1}{L} \ln\left(\frac{I_0}{I}\right) \quad (33)$$

where σ_{ext} is the specific extinction coefficient. σ_{ext} varies slightly with wavelength from about 9.5 m^2/g at 450 nm to about 5 m^2/g at 1000 nm. The value $\sigma_{ext} \approx 8.5$ m^2/g at 550 nm was used in this study.

There are in principle several different physical mechanisms, which deposit smoke aerosol particles suspended in air on the solid surface at different conditions and rates:

- Particles in external force fields:
  - migration due to gravitation
  - electrical migration due to field charging
  - electrical migration due to diffusion charging
- Thermophoresis

Theoretical models

Coagulation model

Smoke particles have a continuous distribution of sizes as discussed above. There is continuous change in the number of particles during the experiment. A dynamic model of smoke particle settling in the chamber including coagulation is proposed for explaining at least qualitatively the temporal behaviour of smoke density during the experiments:

$$\begin{aligned} \frac{dn_1}{dt} + \frac{n_1}{\tau} + \Gamma n_1^2 &= \frac{dn_e}{dt}; \quad 0 \le t \le t_1 \\ \frac{dn_2}{dt} + \frac{n_2}{\tau} + \Gamma n_2^2 &= 0; \quad t \ge t_1 \end{aligned} \quad (34)$$

where n_i is number densities of particles, t_1 is ending time of radiation exposure, Γ is a coagulation rate constant, τ is a time constant for deposition velocity and n_e is number density of particles emanating from cable sample. Curve fitting using equations (34) was tried leading to a fair agreement as shown in figure 84.

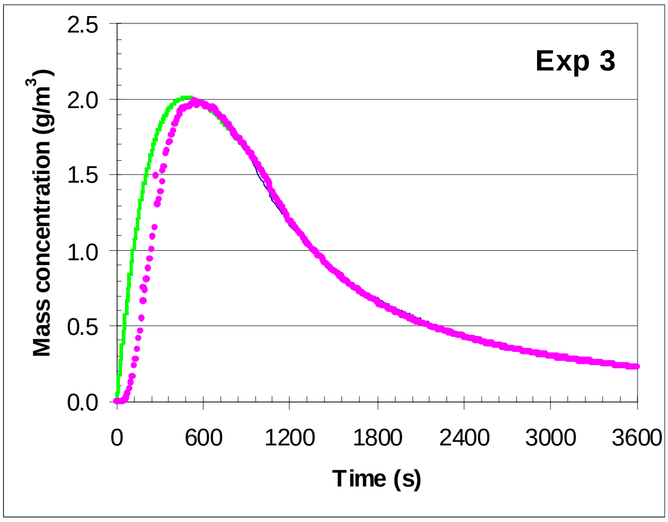


Figure 84. Curve fit (solid) of experiment 3 (dots) to the mass concentration using a simple dynamic model from equations (34).

Soot resistivity model

For understanding temporal behaviour of insulation resistance (figure 87) a simple theoretical model is proposed. Soot deposit is presumed to be a (semi)conducting layer. The deterioration of insulation resistance is caused by ohmic bulk conduction through deposited soot between the copper strips of the comb figure on the board (figure 85).

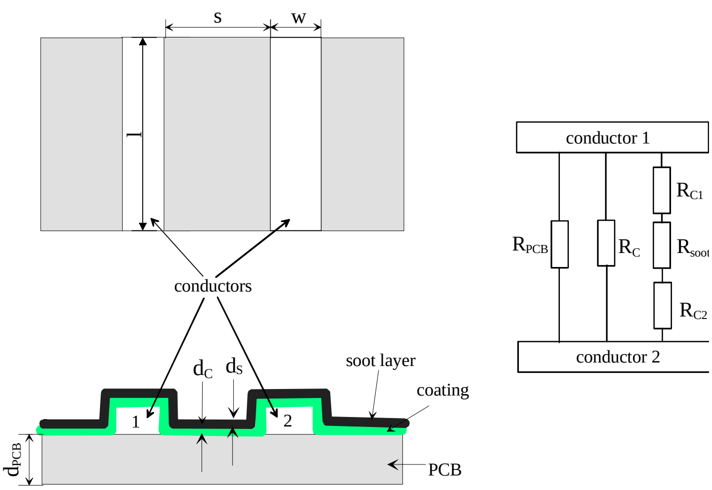


Figure 85. Electrical conduction on a circuit board (PCB) with soot and coating (left) and equivalent circuit (right) with resistances in parallel.

The measured insulation resistance R of the comb figures is the clean printed circuit board insulation resistance R_{PCB} and the soot resistance R_S in parallel. The ratio between soot deposition thickness $d_s(t)$and specific resisitivity ρ_s of soot is then

$$\frac{d_s}{\rho_s} = \frac{s}{nl}\left(\frac{1}{R} - \frac{1}{R_{PCB}}\right) \tag{35}$$

where l is the length of the conductors, s is the constant distance between the conductors and n is the number of soot strips between the conductors.

Considering a situation with two parallel conductors protected with a thin coating (figure 85) the possible paths of conduction and corresponding resistances are

- through the circuit board material, R_{PCB}
- through the layer of coating between the conductors, R_C
- through the coating layer on conductor 1, the soot layer and the coating layer on conductor 2 in series, $R_{C1} + R_S + R_{C2}$

The measured resistance R is then these resistances in parallel

$$\frac{1}{R} = \frac{1}{R_{PCB}} + \frac{1}{R_C} + \frac{1}{R_{C1} + R_s + R_{C2}} = \frac{l\,d_{PCB}}{s\rho_{PCB}} + \frac{l\,d_C}{s\rho_C} + \frac{1}{2\rho_C \dfrac{d_C}{l\,w} + \rho_s \dfrac{s}{ld_s}} \tag{36}$$

where ρ_{PCB} and ρ_C are the resistivities of printed circuit board and coating, d_{PCB} and d_C are thickness of printed circuit board and coating and w is the conductor width.

To obtain a general view on the behaviour of conductivity as expressed by Equation (36) as compared with conductivity of a clean board we calculate the ratio of these conductances denoting it by $\boldsymbol{\gamma}$:

$$\boldsymbol{\gamma = 1 + \frac{\rho_{PCB}}{\rho_C}\left\{\frac{d_C}{d_{PCB}} + \frac{1}{\dfrac{2d_C d_{PCB}}{ws} + \dfrac{\rho_s}{\rho_C}\dfrac{d_{PCB}}{d_s}}\right\}} \tag{37}$$

where all variables are arranged to form nondimensional ratios of two similar quantities. Plotting $\boldsymbol{\gamma}$ as a function of coating thickness using soot thickness as a parameter for given PCB-cards, effect of soot in coated cards can be easily estimated.

Experimental

Four experiments were performed, three experiments with a printed circuit board containing four comb figures and one experiment with an ABB DSAI 130 circuit board similar to those of the NEMO control system used in TVO nuclear power plant.

The circuit board was placed in vertical position in the centre of the smoke chamber and exposed to smoke from a smouldering PVC cable jacket. The cable jacket sample was exposed to thermal radiation from a radiator cone at an irradiance level of 25 kW/m^2 in the absence of a pilot flame. The duration of radiation was 15 min and smoke deposition was allowed in the chamber for the next 45 min, after which the smoke chamber was vented. Recovering of the circuits was monitored for 24 hours after venting.

Insulation resistance of comb patterns

The comb figures had 0.2/0.2 mm, 0.3/0.3 mm, 0.5/0.5 mm, 0.7/0.7 mm line widths and clearances (figure 86). The comb patterns were intentionally without protective coating.

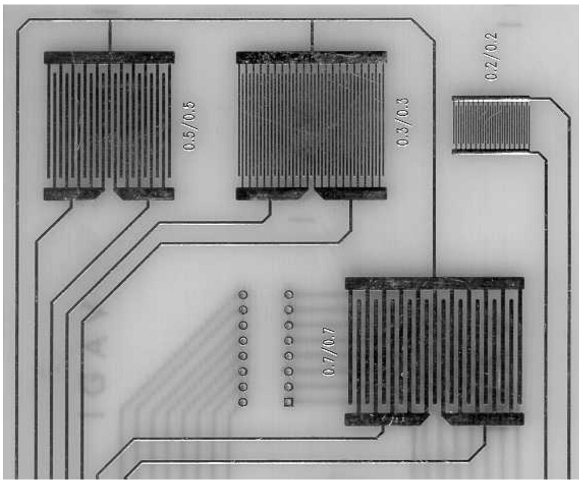


Figure 86. Comb patterns studied in smoke chamber.

Insulation resistance of commercial circuit board

Insulation resistance was measured at three locations between conductive leads and ground. Distance between active and ground leads varied. The minimum distance in all locations was no more than 1 mm.

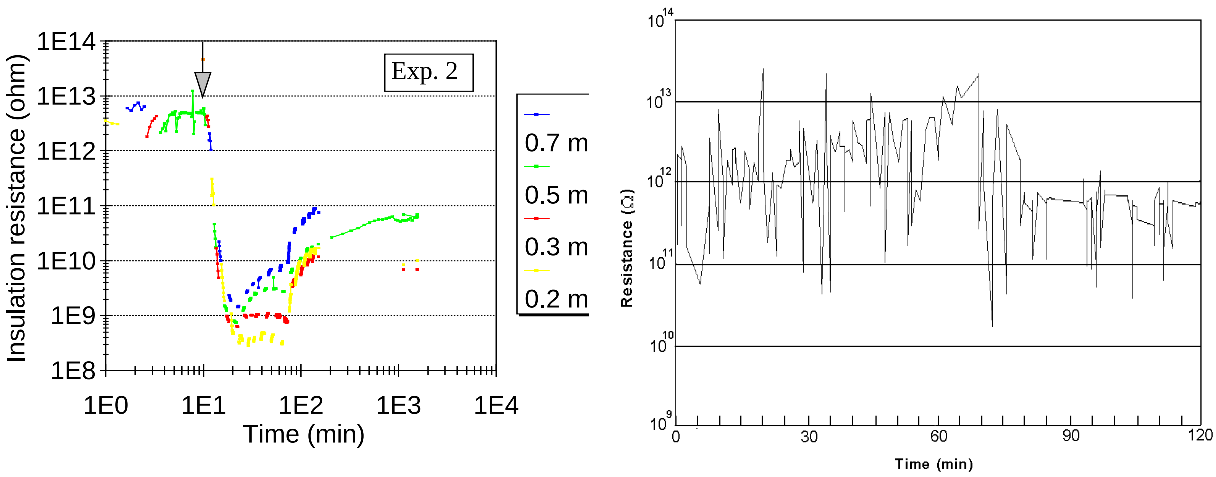
Other measurements

In addition the following measurements were performed:

- Optical density of smoke by monitoring light transmittance through smoke
- Mass deposition of soot on a vertical and a horizontal thin aluminium foil
- Mass of the cable jacket sample
- Gas and surface temperatures
- O_2-, CO_2- and CO gas concentrations.
- Stability of analog input channels
- Time constants of digital electronics.

Results

Insulation resistance of comb patterns is presented in figure 87a and insulation resistance on the commercial circuit board is presented in figure 87b.



....   b)

Figure 87. a) Insulation resistances from comb figures. Start of radiation exposure is indicated with an arrow. b) Insulation resistance on the commercial circuit board. Irradiation of cable sample starts at 5 min.

d/ρ ratios from resistance measurements for the four comb patterns are presented in figure 88. The ***d/ρ***-curves seem to saturate at $1 \cdot 10^{-12}$ S. The average soot deposition in the experiments was 1.0 g/m^2. Assuming a soot density of 1880 kg/m^3 (Krishnan et al. 1999) one obtains a soot deposition thickness of 0.53 μm and an effective soot resistivity of 10^6 Ωm.

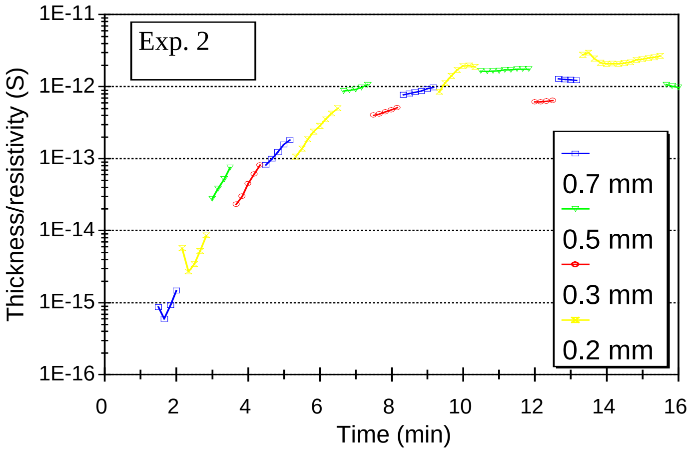


Figure 88. ***d/ρ*** *ratios from resistance measurements for comb patterns.*

Resisitivities of metals are typically of order 10^{-8} Ωm, semiconductors 10^2–10^4 Ωm and insulators 10^{10}–10^{18} Ωm in magnitude. Comparing the effective soot resistivity in the smoke experiments with these values indicates that the conductivity of the deposited soot is poorer than typical semiconductors but better than typical insulators.

The influence of coating is here considered on a level of magnitude study. Material parameters are presented in table 7. Conductor width w = 0.1 mm, clearance between conductors s = 0.1 mm and conductor length l = 10 mm are used as typical dimensions. Calculating numerical values one obtains the different resistances presented in table 7 (cf. Equation 36). It is noticed that the coating resistance effectively hinders leakage of current to the semiconducting soot layer.

Using literature resistivity values for electronic component materials and our measured values for soot, the efficiency of protective coatings can be evaluated. In figure 89 γ according to Equation 37 is plotted for the real PCB used in our experiments.

Table 7. Typical values of resisitivity ρ and thickness d for PCB, coating (from literature) and soot (from measurements) and calculated resistances explained in text.

	ρ (Ωm)	d (m)	Resistance (Ω)
PCB	$10^{12}…10^{13}$	10^{-3}	$R_{PCB} \sim 10^{13}…10^{14}$
Coating	$10^{13}…10^{15}$	10^{-5}	$R_C \sim 10^{16}…10^{18}$ $R_{C1} \sim 10^{14}…10^{16}$
Soot	10^{6}	10^{-6}	$R_s \sim 10^{10}$

It is clearly noted that coating is very efficient to protect against soot deposition and that no effects on insulation resistance deterioration should occur for the real coated circuit studied electrically here. The left-hand parts of these plots are only academic, since the layer thicknesses in real dimensions would be subatomic in size. Even then they show that coating material as an excellent insulator is a very efficient way of protecting the circuits.

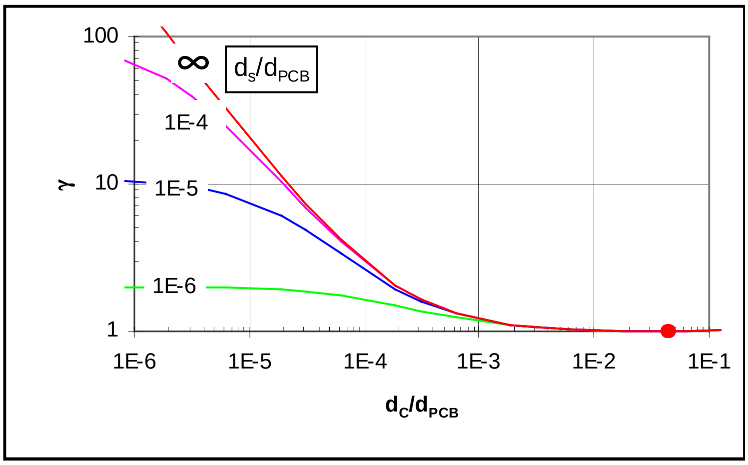


Figure 89. Effect of protective coating on the conductivity of a PCB laden with soot. Dimensions from the real PCB used for experiments. Typical coating thickness 10 μm corresponds to 0.006 on nondimensional horizontal scale.

Conclusions

Smoke exposure experiments on real and mock-up electronics circuitry has been carried out. Special attention was paid on fouling of insulation resistance due to soot layer accumulating from the smoky environment on the circuitry. The major component of soot is carbon. Deposits of soot particles are not expected to be electrically conductive like a graphite layer but rather likely semiconductive. Despite that, the conductance of a soot layer is high compared with conductance of good insulation materials, which are used as boards of printed circuits. Therefore, soot layer on an unprotected printed circuit board deteriorates insulation resistance between metallic electrical conductors.

The soot deposition effect on unprotected circuits is considerable: in the insulation resistance measurements of comb patterns the change is 3–4 orders of magnitude. After ventilation the worst case deviation from the original values was a factor of 10–100 in insulation resistance.

As an example of modern electronics, components on the ABB DSAI 130 analog input board were studied. No significant change in insulation resistance was noticed during the smoke exposure because of protective coating of circuits.

In this paper such instrumentation was attached in a smoke exposure chamber, that dynamic behaviour of smoke emanating from a smouldering PVC source could be monitored continuously. A rough well stirred reactor theory, adapting existing models on smoke agglomeration and settling, was proposed, which supports understanding of the salient dynamic processes of smoke ageing during the exposure.

The model for deterioration of insulation resistance allows a quantitative tool to estimate requirements for protection of electronic circuitry against soot by using various coatings. The model shows as expected, that modern electronics, like that used for today's programmable automation circuits, is in principle more vulnerable for smoke and soot than the conventional analog control circuitry used at the time our nuclear power plants were built. As for insulation resistance our experiments showed that use of protecting coatings could reduce the problem to a tolerable limit. There are still other electrical phenomena, which we did not study here. Therefore, our conclusions from this work are not an unlimited clearance for all smoke related problems on control electronics.

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5.11 PASSI special report

Bayes networks in reliability analysis of software-based automation systems

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VTT Automation

Abstract

Quantitative reliability models of programmable systems are presented. The models are based on Bayes network approach. The models are applied to a simple example case where data from different operational profiles are combined. In addition, the approach is applied to determine the amount of data from another operational profile needed to compensate the lack of data from the operational profile under consideration .

Introduction

Some safety authorities, e.g. the Finnish radiation and nuclear safety authority (STUK), require that the programmable automation systems must be taken into account in plant specific PSAs (see YVL Guide 1997). The quantitative reliability analyses should be either as realistic as possible or sufficiently conservative, which creates a need to analyze and utilize the operating experience.

Characteristic of the reliability estimation of safety critical systems is high reliability requirements with only little statistical evidence available. Thus it is important to utilise all possible evidence in the reliability estimation. Also data from other similar systems and the evidence from qualitative characterisations of the system should be applied. In estimating the reliability, all this evidence should be taken into account in a consistent way. A possibility to do this in such a way that each piece of evidence contributes coherently to the final reliability estimate, is the use of Bayes network models. (see Littlewood & Fenton 1996, and Dahll & Gran 2000).

In this paper, we present how different pieces of evidence can be interpreted as variables of a Bayes network model. Furthermore, we describe an simple example on an application of the model to quantitative analysis of operating experience.

Bayes networks as reliability models of programmable systems

Any Bayesian probability model is a joint probability distribution of all random variables of the model. A Bayes network is a graphical representation of this joint distribution. It is a directed graph, in which all nodes correspond to random variables, and directed arcs correspond to the dependencies between these variables. The joint probability distribution of a Bayes network is constructed by postulating the conditional distributions of each variable given its nearest parents.

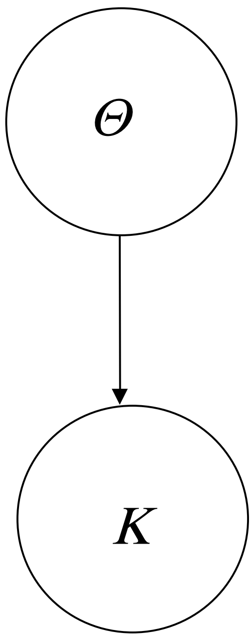
Often Bayes networks are hierarchic, connecting the observable variables with parameters and other unobservable variables. The conditional probability distributions between the variables determine how the uncertainty about the values of hidden variables can be reduced by observations. This is done by using the Bayes theorem, which yields the conditional distributions of hidden variables given the observed evidence.

In the reliability analysis of programmable systems, the hidden variables of interest are, for example, the failure probability of the system or the number of errors in the systems software. They are dependent on other, possibly observable variables describing the structure of the software, its development process, or operating experience.

As the simplest case we model the case with have some operating experience from a single programmable system. We have observed that the system has been demanded n times and k failures have occurred. In addition to this evidence, we have some information on the systems structure and development process. The latter evidence is used to construct a prior distribution for the systems failure probability parameter θ. In principle, we could model also this ”prior information” more explicitly as a Bayes network. Here we assume that this information is contained by the prior distribution $p(\theta)$.

The number of failures, K, in n demands is a random variable modeled with a binomial distribution, conditioned on the failure probability parameter θ. The prior distribution of θ is the logit-normal distribution, due to its flexibility in more complex modeling cases. The Bayes network corresponding to this simple model is in figure 90.

Often operating experience is available from similar programmable systems, which operate in different kind of environments. Due to different environment, the input sequences to the system are different, and the hidden software errors are not revealed with same frequency in all similar systems. This means that the failure probability, which is dependent on the environment, is not same for all similar systems. However, it is clear that the data from all systems is useful evidence on the reliability of the specific system under consideration. To describe this situation, we assume that there is a common failure parameter, which determines partly the failure parameter of the system under consideration and that of other systems. The differences between the systems are modeled as additive random disturbance to the common parameters. Depending on the statistical spread of the disturbance, the information from other systems has different weight on the final reliability estimate of the system under consideration. In our case, the statistical disturbance is related to the differences of operational profiles and design features of the systems. We denote the failure probability of the system under



$$p(\theta): \ln\left(\frac{\theta}{1-\theta}\right) \sim N(\mu, \sigma)$$

$$P(K = k \mid \theta) = \binom{n}{k} \theta^k (1-\theta)^{n-k}$$

Figure 90. Simple basic model.

consideration by θ, and the those of other systems by θ_I. They depend on each other through logit-transform and an additive, normally distributed disturbance describing the difference of the operational profiles. The model is illustrated in figure 91.

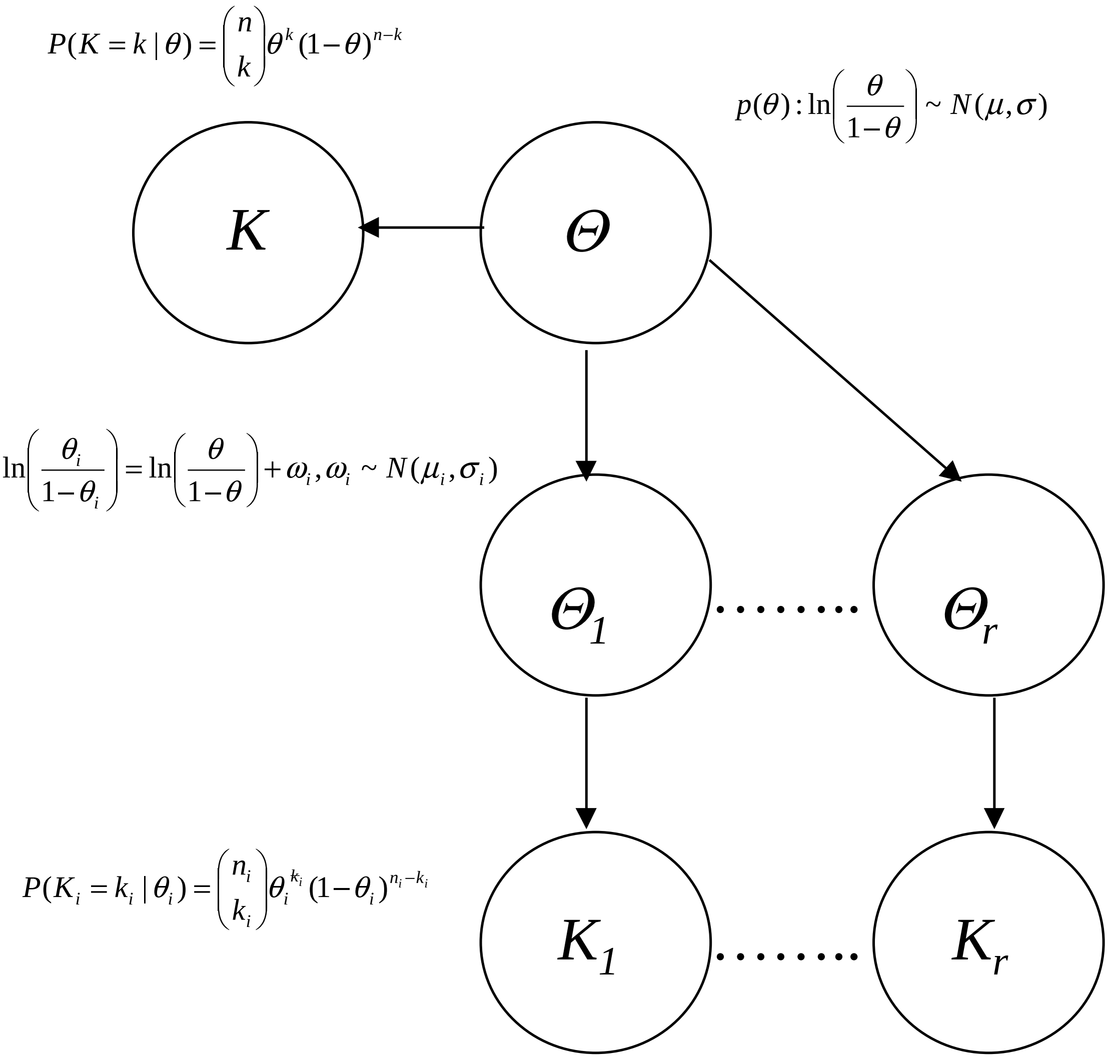


Figure 91. Model for data from different operational profiles.

The information on the reliability of the system under consideration is include in the posterior distribution of θ, given the observed numbers of failures. In our case, it is not possible to determine the posterior distribution analytically, and we apply the Gibbs sampler algorithm, see e.g. Gilks et al. (1996).

The modeling approach makes it also possible to use the evidence from earlier versions of the same software-based system. Since the successive versions often use partly common software, it is evident that the failure parameters of the versions are dependent on each other. By assuming a common failure parameter, on which the failure parameters depend, we can build a model in the same way as above. Similarly, it is possible to utilize evidence from different systems, which are based on the same system platform.

An example on the application of the model

An interesting question, when combining evidence for a same system functioning in two different operational profiles, is how much the evidence coming from the other operational profile can compensate the lack of operational experience obtained from different operational profile (figure 92).

We assume that the logit-prior distribution of θ has parameters $\mu = -4.6$ and $\sigma = 1.0$, which correspond to the prior median $1.0 \cdot 10^{-2}$, and the prior *97.5*-percentile $6.7 \cdot 10^{-2}$. We calculate how many successful demands from a different operational profile are needed to compensate the lack of evidence from the system under consideration. We start from the situation, in which *1000* successful demands of the system under consideration are observed. Next we decrease the number of the successful demands of our system, and determine the number of successful demands from another operational profile which should be added to the evidence in order to obtain the same *97.5*-percentile as in the case with *1000* demands from original profile. We do this for three different values of variance of the disturbance parameter.

We note that although the variance of the additive disturbance is small $(\sigma = 1.0)$ quite large amount of evidence is needed to compensate the reduction of evidence from the original profile. When the variance of the additive disturbance grows, the amount of additional evidence needed to compensate the reduction grows rapidly.

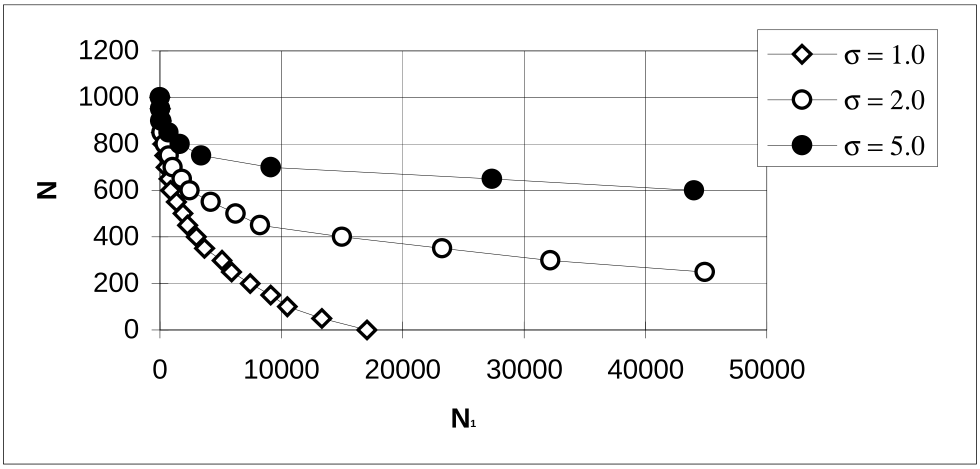


Figure 92. Compensating the lack of evidence with data from another operational profile. (N= number of successful demands from original profile, N_1= number of successful demands from the other profile.

Conclusions

The Bayes networks provide an efficient way of combining all kind of evidence together, and thus generating estimates for the reliability of software-based digital systems. The advantage of using such a representation is that it makes the modeling easier and the model more transparent.

Since the statistical evidence is often insufficient, it is important to use all possible data from different sources. Here we present an example application of the model to a case where operating experience from different operational profiles is combined with a system specific data. The transparent Bayes network model describes this situation explicitly, and thus helps in evaluating the weight of each piece of evidence. The same idea applies also to the situations, where the operational experience from earlier versions of same system.

As a statistical model, Bayes network is some kind of black-box approach to software system reliability modelling. It does not directly take into account the information of systems internal structure, which definitely has an impact on the systems reliability. In the Bayes network approach, this kind of features are included in the prior distributions.

Although the approach is developed for analyzing software-based systems, it is equally applicable to any case in which there are data from many sources

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5.12 METRI special report

Application of expert panel approach to support RI-ISI evaluation

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Abstract

The paper describes the expert panel methodology developed for helping risk-informed decision making, and its application to evaluate the preliminary results in the STUK's RI-ISI (Risk-Informed In-Service Inspection) pilot study. The aim of an expert panel is to achieve a balanced utilisation of information and expertise from several disciplines in decision-making including probabilistic safety assessment as one decision criterion. In the RI-ISI application, the expert panel approach was used to combine the deterministic information on degradation mechanisms and probabilistic information on pipe break consequences. The expert panel served both as a critical review of the preliminary results and as a decision support for the final definition of risk categories of piping.

Introduction

Regulatory bodies and nuclear utilities are increasingly interested to use the probabilistic safety assessment (PSA) results in decision making related to the operation and maintenance of nuclear power plants. Experiences show that risk-informed applications may lead to a simultaneous icrease of safety and decrease of costs. When moving towards risk-informed regulation and plant operation, decision-making situations often involve expert opinions from several disciplines.

This paper describes an expert panel approach developed for decision-making situations where probabilistic and deterministic information from various sources should be combined in order to achieve an optimal decision. The approach was applied to the pilot study on risk informed in-service inspection (RI-ISI), conducted at STUK during 1998-1999 (Mononen et al. 2000). The aim of the study was to test the suitability of EPRI RI-ISI methodology (Gosselin 1997) for helping regulatory decision making in Finland. In this application, the expert panel approach was used to assess the preliminary categorization of piping segments of selected systems from Olkiluoto and Loviisa power plants.

The panel consisted of experts representing several areas such as process, structural and material engineering, in-service inspections, and PSA. Further, two external normative experts conducted the discussions and acted as facilitators in the panel. The aim of the panel was to discuss the justification behind the categorization of segments, and to identify related uncertainties. The preliminary categorization was verified and changes in segmentation and categorization were identified. The panel process can be considered both as quality assurance of the results and as a decision support for the final categorization of segments. The paper summarizes the experiences from the expert panel application to the RI-ISI case study.

Description of the Expert Panel Approach

Planning of activities like in-service inspection program or selection of accident management measures can be seen as a complex decision problem. In making such decisions, various kinds of expertise together with subjective assesments are utilised in order to evaluate the decision alternative against many, possibly conflicting criteria. In nuclear safety decisions, the criteria may be related to risks, measured e.g. by quantitative PSA results, consequences of accidents, properties of the process and materials, and costs. In some cases, the decision criteria are conflicting, and there are no decisions, which are best with respect to each decision objective. An important aim of the decisions is to find cost-effective approaches for improving and maintaining the safe operation.

Decision analysis is an approach for resolving difficult decision situations. It aims at modeling the subjective assessments of the decision-maker. During the course of a decision analysis, it is essential to distinguish decision goals and attributes, and the uncertainties related to the state of the nature and the outcomes of decisions. In other words, facts, their uncertainties and values related to the situation are identified. In addition to this, establishing a clear structure for the decision problem in hand is important.

A complete decision analysis requires also the construction of quantitative decision model for identifying the solution, which fulfils the goals and values of the decision-maker. From a practical point of view, a straightforward application of formal decision analysis is often too resource consuming, as the earlier experiments have shown (see e.g. (Holmberg et al. 1994)). However, in risk-informed decision making, a suitable format for group decision making is needed in order to structure the problem and find a balance between the risk-based and other criteria. It can also be expected that the discussion between experts reveal contradicting criteria, which can not be treated without a systematic approach.

In the following description of the approach, we assume that the technical experts have already been selected, and they have performed their analyses and formed their initial opinions on the subject. The role of the expert panel is to synthesize the views of various experts and identify and characterize the uncertainties in their analyses in order to find a balance and consensus between the possibly contradicting arguments of experts representing different disciplines. Further, the panel sessions serve as a tool for revealing new aspects that experts would not have considered without communicating together in a structured discussion.

To achieve a good basis for decisions, a decision-maker, a referendary (e.g. project leader), technical experts, and normative expert(s) should participate in the experts panels. The role of these participants depends both on the case at hand and the resources available for discussions and additional work. The expert panel process has the following basic steps: 1) structuring of the problem, 2) development of suitable formats for identifying the background for experts' judgements and related uncertainties, 3) preparation for the panel session, 4) panel session(s), and 5) reporting of the results. If in the course of the panel session, needs for additional analyses or checks are identified, tasks are assigned for responsible experts and a new panel session is arranged after obtaining the complementary information. The roles of various participants in the steps of an typical expert panel process are described in table 8.

Table 8. Role of participants in an expert panel process.

	decision maker	**referendary**	**technical experts**	**normative expert**
problem structuring	presents the strategic view, role of the decision	describes the case	(give detailed information if needed) [1]	familiarizes with the case, structures the problem
development of formats			(comment the formats)	develops formats based on problem structuring
preparation for panel session		prepares for presenting the case	fill the formats, summarise tehir own analyses	prepares for leading discussions
panel session	(observer, may take part in discussions)	presents the case, takes part in discussions	present their analyses, participate in discussions	leads discussions, facilitates communication between experts, takes notes
reporting		comments and accepts the summary report	(comments the summary report)	summarises the discussions and results of the panel

1. optional participation in parentheses.

Application to RI-ISI

In Risk-Informed In-Service Inspection approach segments of piping are assigned into risk categories based on the probability of a pipe break occurring in the piping and the consequences of a break in that segment. Probability of a break is evaluated qualitatively on the bases of an assessment of the susceptibility of the piping to the degradation mechanisms known to effect such piping. The consequence of a pipe break is assessed by using the PSA model. The principal idea is to redefine the in-service inspection programme according to the risk importance of piping segments. The risk categories are presented in table 9.

Table 9. Risk matrix for pipe segments (ASME 1997).

RISK CATEGORIES: ***LOW*** ***MEDIUM*** ***HIGH***		CONSEQUENCE CATEGORY			
		NONE	LOW	MEDIUM	HIGH
DEGRADATION CATEGORY	LARGE	***LOW***	***MEDIUM***	***HIGH***	***HIGH***
	SMALL	***LOW***	***LOW***	***MEDIUM***	***HIGH***
	NONE	***LOW***	***LOW***	***LOW***	***MEDIUM***

In the STUK-RI-ISI case study, two systems from both the Olkiluoto and Loviisa nuclear power plants were selected for evaluation. The systems were the high pressure injection system at Loviisa and the shutdown cooling system at Olkiluoto plant that are included in the present ASME programme, and the Loviisa emergency feed water system and the Olkiluoto service water system that are not covered by the ASME programme. In the evalution, the piping of these systems were preliminary divided into segments and categorized. Although expertise from structural and material engineering, in-service inspections and PSA was the basis for this preliminary evaluation, there was initially no systematic interaction between the experts from various disciplines.

The expert panel methodology was thus used to facilitate a structured re-evaluation of the preliminary segmentation and categorization, and to achieve a balanced utilization of deterministic information on degradation mechanisms and probabilistic information on pipe break consequences.

In this pilot study, the only decision made was the categorization of piping segments of certain nuclear power plan systems. However, the expertise from several fields and the uncertainties involved in both evaluation of degradation potential and consequences of pipe failures make the problem complicated enough for application of simplified decision analytic methods.

The panel consisted of STUK's experts having knowledge of structural and material engineering, in-service inspections, plant processes and PSA. Further, two external normative experts conducted the discussions and acted as facilitators in the panel. The aim of the panel was to discuss the justification behind the categorization of segments, and to identify possible needs for changes in the original segmentation and categorization.

Specific forms were developed to collect in a condensed form the background information related to the categorization of segments, and to the identify criteria and uncertainties related to the categorization. Following information needs for the review of the degradation potential classification were defined:

- *Description of possible degradation mechanism(s):* sensitivity for conditions, and possible knowledge on degradation rate
- *Influencing factors and their impact in the segment:* material properties, environmental stresses, transient history, geometry
- *Current inspection program and method:* accessibility, inspection method, earlier inspection results, limiting/ restricting factors, worker safety
- *resulted degradation category.*

For the review of consequence categorization, following questions were addressed:

- *Consequences of pipe failure and their models in PSA:* initiating event, CCI standby failure, demand failure, isolation of the leakage, degree of detail of PSA models
- *Uncertainties related to the conditional core damage probability (CCDP) quantification, CCDP estimate*
- *resulted consequence category.*

The experts were provided with the forms in good time prior to the panel, so that they could collect the necessary information. During the panel discussion, each segment was evaluated separately, and the experts were requested to identify the related degradation mechanisms and uncertainties related to the environmental conditions for each segment. In this connection, the existing in service inspections were reviewed and the factors influencing the effectiveness of inspection were discussed. The consequence evaluation was made by using the plant specific PSAs, and the assumptions and simplifications of the quantitative evaluation were considered in the panel.

Experiences from expert panels

One of the major benefits of the expert panels was the identification of needs for complementary information for justifying the categorization of segments. Some of these needs were quite generic and should be taken into account in possible new applications of the RI-ISI methodology. Other more specific needs were related to the analysis of systems in the pilot study.

As the experts were prepared to present their analyses to each other, the insights of various disciplines could be combined in a most useful way. For instance, within such a limited pilot study, the consequence evaluation did not consider the possible secondary effects of pipe breaks in detail. During the panel discussions, insights from process and material engineers helped in identifying and evaluating the most important secondary effects, e.g. impact of floodings or loose piping on nearby equipment. As another example, isolation valve failure probabilities used in the PSA were subjected to criticisms in case of abnormal conditions due to a break in the pipeline.

The panel discussions resulted also in practical recommendations for the plants. For instance, related to one segment, the correspondence of the pressure between testing and demand situations in the testing line were discussed in the panel. It was noticed that the pressure in demand situations of the system is probably significantly higher than during the test, and thus the risk for pipe failure is larger in demands. This has also an impact on PSA calculations where the CCDP is calculated basically with the real test interval. A leakage test was recommended in order to decrease the pipe break risk on demand.

The generic needs of background material that shoud be collected and analysed during the preparation of data for the segment categorization were discussed. The panel process indicated that attention should have been given to the results of previous in-service inspections, which were not analysed in this pilot study. Also, the initial information should contain results of pre-operational system and component tests.

The segmentation according to the degradation potential was discussed in the panels. There were first some proposals to distinguish some individual welds as own segments. However, later on it was decided that no separation is needed because the original segmentation was considered sufficient according to the ASME code case N-578 (ASME 1997). Instead, within some segments, certain locations were identified as most important to be inspected. The segments were categorized conservatively if the information available was considered insufficient to reliably justify a lower category. In several cases, however, it was agreed that the degradation classification of segments could later be lowered depending on further investigations or e.g. results of additional vibration or temperature measurements.

The quantification of pipe break consequences with PSA analyses was problematic in some cases. One difficulty arose from the pipe break modeling of the Loviisa PSA, where all LOCA initiating events were assumed to occur in one redundancy of the plant piping. Further, the consequences could not be straightforwardly evaluated with the conditional core damage probability in the case of a system where a pipe break does not generally cause an initiating event. Although leakages in the system do not cause initiating events, they may have an impact on consequences if an initiating event requires the operation of this system. Thus, the CCDP was conditioned both on the occurrence of another initiating event and a pipe failure in this system. The approximate (and conservative) plant specific PSA-models prevented the exact calculation of these CCDPs, and some of the quantitative estimates could be seen only as relative indications of consequences of pipe breaks. Expert judgement was needed in determining the consequence categories, and this was discussed in the panel.

The panel discussed the categorization adopted from the EPRI approach. The resolution with the consequence categories was felt too low in many cases. Further, in some cases the simple conditioning by the LOCA-initiating event was difficult and made the absolute evaluation impossible. To find a better categorisation rule, some sensitivity studies or even additional research may be needed. Another possibility is to determine the categorisation principles on case by case basis. This requires, however, calibration with other cases. Concerning the degradation categories, it was suggested that the division into four categories instead of three should be considered.

In the course of the panel sessions and after obtaining the complementary information requested by the panel, some changes in segmentation and categorization were made. The panel proved to be an essential part of the RI-ISI pilot study, as it clarified the justifications behind the final risk categories and helped in reviewing and reporting the results of analyses.

Conclusions

The application of risk informed principles is a trial to use PSA together with deterministic analyses in making safety related decisions. This requires not only the straightforward comparison of quantitative risk estimates and results of deterministic calculations, but also a structured decision analytic view on the problem at hand, and balanced combination of expertise from several technical areas. In addition to this, also the impact of related uncertainties must be evaluated. In this paper, a simplified decision analytic procedure for resolving the above issues, based on expert panel approach, is discussed.

The developed expert panel approach was applied to the pilot study on risk informed in-service inspection (RI-ISI), conducted at STUK. The approach enabled a structured discussion between experts from several disciplines, which was felt very useful. Although no clearly contradicting opinions between the experts were arisen, the panels were seen important as quality assurance audits. This was due to careful evaluation of judgements by experts as well as to the emphasis of identifying uncertainties in different kind of analyses.

Along with the increased adoption of risk informed principles, the use of decision analytic approaches may be inevitable. Simplified procedures, such as the expert panels discussed in this paper could help in defining more accomplished methods, which could also be applied to other cases with more challenging multi-criteria decision-making situations with conflicting opinions and criteria.

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5.13 WOPS special report

NPP fire situations: Points of view related to human actions

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Abstract

Nuclear power plants create a demanding environment for fire fighting and simultaneous process control. The special features of fire situations complicate on-line decision making. They are also problematic from the viewpoint of probabilistic fire analysis. The paper introduces a study, the general goal of which, on one hand, is to enhance knowledge concerning the demands of the control of fire situations and, on the other, to support the human reliability analysis (HRA) relating to NPP fires. The aim of the current phase of the study, described in the paper, has been to gain, with the help of expert interviews, more understanding of the nature and problems of NPP fire situations and of the organizational support of the control of fire situations. In addition, characteristic problems and the way of taking account of human aspects in carrying out the probabilistic safety assessment (PSA) related to fire have been discussed with the interviewees. The experts, 15 persons who represented Loviisa and Olkiluoto nuclear power plants, Radiation Protection Centre (STUK) and Technical Research Centre of Finland (VTT), had different perspectives on fire situations and probabilistic fire analysis. They were representatives of control room shift supervisors, simulator instructors, plant and communal fire brigades, plant fire experts and experts of deterministic fire analysis and probabilistic fire safety assessment. The interviews were designed and carried out on the basis of the integration of psychology and reliability analysis. The paper introduces a summary of the preliminary results of the interviews by presenting the experts' points of view on a general level.

Introduction

Fire risk analyses have shown that fires can be a major contributor to nuclear power plant risk (OECD 2000). They are demanding for the control room operators who have to gain an understanding of the whole situation and make operational decisions. Moreover, due to the division of labor in fires situations, the operators have to change

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[5] Pekka Pyy from FINNUS/METRI has participated in the interviews and given advice during writing this paper, which is duly acknowledged.

information and negotiate with the plant fire brigade and, in some cases, also with the communal fire brigade from outside. These circumstances deviate from the conditions of less distributed disturbances.

There are considerable uncertainties in the results of the fire related probabilistic safety assessments (PSA), one of which being related to the modeling of operator responses to the fire (OECD 2000). In fire PSA, the main assumptions are brought into accord with the suppositions on PSA concerning the internal events (transients and LOCAs). Sometimes, the models have to be extended in order to achieve the detail level and coverage required by the fire situation. The extension of the internal event PSA is, however, problematic because fire assessments require a lot of very detailed layout data, not typically included in the former, like knowledge of the location of the cables in the plant. The question of the applicability of the human error probabilities in the fire scenarios is important and has to be raised up (Kazarians 1999). According to e.g. Gil & Yllera (1999), human reliability methodology becomes more uncertain when the analyses applied have been adopted from other parts of the PSA.

There seems to be a need for considering NPP fire situations from a wider perspective, by trying to take the human aspects more comprehensively into account. The FINNUS strategy group, for its part, has expressed the need to study fire related HRA. The study presented in this paper is a part of a subproject belonging to the WOPS project (Working Practices and Safety Culture in NPP operations). The subproject, the theme of which is Human reliability in fire situations, is aimed to support fire related HRA by enhancing knowledge of the demands of the control of fire situations. The study is carried out together with the METRI project (Methods of Risk Assessment), by integrating the perspectives of psychology and reliability analysis. Also the co-operation with the FISRE project (Fire Safety Research) has been established and will be discussed in the next phase of the study.

The focus of the study presented here has been on NPP fire situations as the context of decision making of the different actors, i.e. control room operators and fire fighters. The paper introduces a summary of the preliminary results. A more comprehensive description of the study and the results will be presented in a forthcoming report. The need for a special HRA for fire situations will be considered in the next phase of the study (see chapter 4).

Goals and method

The goal of the current phase of the study has been to gain, with the help of expert interviews, more understanding of the nature and problems of NPP fire situations and of the organizational support of the control of fire situations. In addition, characteristic problems and the way of taking account of the human aspects in carrying out fire PSA have been discussed with the interviewees.

The task was to find out

1) what kind of problems the control room staff and the fire fighters may encounter in the control of fire situations and in what ways these problems complicate decision making

2) what kind of developmental needs there are relating to the organizational support for the control of fire situations, i.e. to procedures and training

3) what kind of difficulties the analysts have encountered and in what way the human aspect has been taken into account in carrying out fire PSA

In the approach the problems were emphasized because the factors related to them help to make visible the essential demands of the activity of the different actors involved in the situation. Therefore problem oriented interviews were used to elicit opinions of experts representing different views on fire situations. The approach has been constructed and applied in earlier studies concerning conventional power plants, nuclear power plants and paper making (e.g. Hukki 2000). The origins of the approach are in the contextual psychological approach developed in VTT Automation (see e.g. Hukki & Norros 1993, 1998, Klemola & Norros 1997, Holmberg et al. 1999, Norros & Nuutinen, submitted).

The experts, 15 persons who represented Loviisa and Olkiluoto nuclear power plants, Radiation Protection Centre (STUK) and Technical Research Centre of Finland (VTT), had different perspectives on fire situations and fire analysis. They were representatives of control room shift supervisors, simulator instructors, plant and communal fire brigades, plant fire experts and experts in deterministic fire analysis and probabilistic fire safety assessment. The interviews of the representatives of the communal fire brigades are not included in the results introduced in this paper but will be presented in the forthcoming report.

The interviews were designed and carried out on the basis of the integration of psychology and reliability analysis. They consisted of questions belonging to the following areas: a) characteristic features of NPP fire situations, b) problems of fire situations from the fire control and process control point of view and c) problems of procedures and training and d) problems of fire PSA (and also deterministic fire analysis). The intention was to construct a many-sided view on fire situations in order to outline the demands of decision making on a general level. The idea was to find and make visible points of view which can serve as a basis for a more detailed conceptualization of the demands of control of fire situations in the next phase of the study (see Discussion). Another way of utilizing them would be the development of procedures and training in the plants.

The expressed opinions concerning the problems of decision making were analyzed by classifying them according to the following themes:

- the type of questions the control room operators and the fire fighters have to form an understanding of, and the type of operational decisions they have to make

- what kind of factors may impede or complicate the formation of understanding and the making of operational decisions

On the basis of the classified opinions the general outline of the characteristic demands of decision making and of the possibilities to fulfill them was made. The focus was on the effects of the fire on the event identification and on the evaluation of the operational possibilities. Decision making was analyzed also from the point of view of roles and co-operation of the different actors (i.e. control room, field operators and fire fighters) involved. In the final phase of the analysis, the points of view concerning the problems of decision making, the organizational support and the characteristic problems of and the way of modeling human actions in fire PSA were considered in connection with each other.

Results

Problems of decision making in the control of fire situations

In the following the interviewees' points of view concerning the control of fire situations and the organizational support for decision making are shortly outlined in the form of a collage. The encountered problems are described concentrating on the main points.

Process control and simultaneous fire fighting

The most characteristic features of fires are *uncertainty* and *vagueness* which make it difficult to fulfill the demands of process control and fire fighting. The consequences of the fire can range from insignificant to destructive and that is, for the most part, due to the difficulty to foresee the propagation of the fire and the effects of the fire on the plant process and the systems. The effects can be indirect and unexpected. For example a large fire may have no effect on controllability of the process and a small one in the control room may lead to severe consequences. When compared with failures of devices in process disturbances, fires generate a lot of restrictions to activity which cannot be anticipated. In addition to the effects on the information and on the equipment also other factors like smoke, heat, poor visibility, toxic gases etc. may worsen task performance. As a result of these factors the control of the situation may be, at worst, extremely difficult.

A significant aspect of fire situations is their *complexity* from the decision making point of view. The most complicated seem to be cable fires, the probability of which is small but the consequences may be severe. Cable fires and fires in the instrumentation cubicles and in the cable spreading room can be very difficult to handle because fire may reduce the availability of information and of safety related equipment. They may also cause inadvertent instrumentation and control signals resulting to misleading control room information, spurious component starts of equipment and dysfunctioning of automatic control systems of the plant.

In the following outline the problems encountered in fire situations are described, firstly, from the process control point of view and, secondly, by taking into account the whole situation, i.e. the process control and the simultaneous fire fighting.

When taking care of the *process control* the operators should be able to gain understanding of the situation in the sense of what has happened and what are the operational possibilities. Evaluation of the situation is always deficient in the early phases of the fire. Diagnosing may be impaired due to reduced or misleading information The operators should be able to find out if there is a fire or not, and in the case there is one, to locate it and to identify a) how serious it is, b) what are the dimensions of the fire and c) what are the prospects of the propagation.

It is very difficult to the operators to evaluate the effects of the fire on the process. If the control room information and the state of the process do not match the evaluation is much more difficult. The ability to evaluate the reliability of a measurement is important. There are redundant ways to find out the reliability of important measurements and one has to compare different sources of information with each other. In addition, a field operator can be sent to the plant to get direct information, and the fire fighters and other persons in the plant can tell about their perceptions.

The automatic protection systems are based on measurements. If the latter ones are lost, the operators do not know if the controls are functioning adequately or not. The ability to identify lost equipment and to anticipate what devices and systems are still functioning is important. Knowledge of cables and instruments located in a certain room and of routes which are critical in the sense of nuclear safety is needed here. A "process feel” helps to orientate in the situation. Operationally complicated situations may, furthermore, lead to shortage of personnel because the control operations may demand participation of many operators. The circumstances may also demand wearing of protective equipment.

When looking at the problems from the viewpoint of the *whole situation* the scene is the following. The shift supervisor has to take an extra role in the early phases of the fire situation. He has the responsibility of both fire fighting and process control at the same time. If the communal fire brigade has been alarmed, it means that the fire fighters on site represent two types in the following sense. The plant fire brigade knows the plant quite well but consists only of a couple of men. There are much more members in the communal fire brigade but they know the plant very poorly. After the communal fire brigade has arrived its chief takes over the responsibility. Evaluation of the operational possibilities is not, however, easy for him, due to lack of knowledge of the plant and the process. The fire brigade of the plant can help in this respect to a certain extent but it is necessary to get information from the control room. Normally, a field operator who knows the plant well is sent to serve as a guide when the communal fire fighters enter the plant. In addition, the fire chief is in contact with the shift supervisor in order to change information concerning the fire and to get advice about the effects of the fire on the plant. The operational decisions are made together and the control room staff has to perform their tasks in a way which makes it possible to the fire fighters to do their job. Due to these circumstances, the way the shift supervisor is able to describe the situation from the process point of view is very important. The terminology he, and also the field operator, uses is not familiar to the fire chief and this may easily lead to misunderstandings. In addition, the contact with the shift supervisor may sometimes fail, due to disturbances in telecommunications, which means that the fire chief may have to make the decisions on his own.

On the basis of the interviewees’ points of views, described above, the problems encountered in process control and simultaneous fire fighting seem to be related at least to the following issues 1) the nature of the fire itself, 2) the changes of the roles of the control room staff and 3) the division of labor and communication between the different parts (the control room operators and the fire fighters) involved in the control of the situation. In the following the some central demands of the decision making of the different actors and their possibilities to fulfill these demands in complex fire situations are classified according to the discussed aspects.

Problems related to the fire itself

Control room staff

- should find out what has happened (existence, location and severity of the fire and the state of the process) but the information may be deficient or unreliable
- should make operational decisions but it is difficult to anticipate the effects of fire on the devices and systems
- should make operations, e.g. a shutdown, but the necessary systems may not function
- loss of information concerning important systems may lead to a situation where the process cannot be controlled although it is completely functioning

Fire fighters

- should make decisions but it is difficult to anticipate the way of fire propagation

Fire fighters & control room staff

- very effective task performance is needed but the circumstances (time pressure, poor visibility, possibility of toxic gases etc.) cause worsened capacity, stress and even fear

Problems related to the roles of the actors

Control room staff

- the shift supervisor should form an understanding of the state of the process but it is difficult to monitor the process because of the tasks relating to the additional responsibility of fire fighting in the early phases of the fire situation
- the shift supervisor should form an understanding of the whole fire situation but it is difficult because he has simultaneously two roles and time pressure in the early phases of the fire situation

Problems related to the division of labor and communication

Fire fighters

- should form an understanding of the best strategy of extinguishing the fire but they know the plant poorly

- should form an understanding of the effects of the fire and of fire fighting on the process but do not know the process
- need information and advice from the control room but the operators' terminology is not familiar
- need information and advice from the control room but may be too busy to be able to pay attention to the given information

Developmental needs of procedures and training

The ability to act adequately in fire situations can be enhanced by appropriate organizational support in the form of procedures and training. A short summary of the interviewees' points of view concerning this support is presented in the following.

As to the *control room* the procedures may be too reduced, open to various interpretations. In fire situations there may be moments when there is not time to resort to the procedures. The essential knowledge must also be "in the head". This is also due to the fact that some types of fires, e.g. cable fires, are so complicated that it is impossible to develop procedures for them.

Development of operator training is very important. Rehearsing with the plant simulator is not sufficient due to deficient realism, e.g. the reduced possibility to send a field operator to the plant. Besides, development of fire scenarios for the simulator is very laborious. There should be training of strategies concerning acting in fire situations. Training of fire situations is not, however, considered as important as training of process disturbances due to the vagueness of the former.

Another issue is that the operators should know the results of risk analyses concerning the plant. They should have knowledge of the fire scenarios used in the analyses and of the critical issues concerning them. Also relevant results of the deterministic analysis should be available, e.g. knowledge concerning the reliability of information of valve actuators as a function of time elapsed after the ignition of the fire.

It is not easy to develop procedures for *fire fighting* in nuclear power plants. Conventional fire guidelines are fragmentary and do not apply to nuclear power plants. The current procedures may be too complicated or, on the contrary, too reduced check-lists, the optimum being in-between. They may also be out-of-date. An extra problem is that after the fire alarm has come it is difficult to have time to look at the procedures because the time to get ready for fire fighting is very short.

The firemen lack personal experience of real NPP fire situations, and it is difficult to form an understanding without it. It is, however, difficult to rehearse fire situations due to insufficient realism. Training should be developed because its significance is prominent. A way to improve the situation would be that the most critical targets of the plant would be defined for making room or system specific plans for extinguishing the fire. These plans should be utilized in training.

There are developmental needs as to the feedback of the experience of NPP fires to the different actors involved in fire situations. The experience from other plants may not, however, be very useful because the plants may differ too much. This is the case especially with the small incidents. The fire scenarios concerning big events are usually known and utilized in training.

Problems related to probabilistic fire safety assessment

The interviewees' points of view concerning probabilistic fire safety assessments are shortly introduced in the following.

Probabilistic fire safety assessment (fire PSA) will become more and more important because many safety related improvements have been already made, on deterministic basis. A big problem in the PSA is the fact that the effect of all different types of fires cannot be defined. There are many interdependencies in nuclear power plants and they are difficult to comprehend. Therefore there is no way to develop any straightforward conceptualization. The simplicity of the fire PSA model, especially the limitations included, may be regarded as a problem. It is a consequence of the attempts to secure the functioning of the model and/or to diminish the work load. Also the shortage of statistical data is a problem in the fire PSA. The analyses should, however, be plant specific because generic experience does not necessarily help in the case of an individual plant. When the knowledge is missing, one has to make conservative, i.e. pessimistic assumptions. For example the location of the cables is not always known and if it would, one could not, however, know what would be the order of the failure and the failure mode (e.g. short cut or loss of supply); the worst possible combination is always supposed. Still, the conservative stance may not be the right solution, a better way being collecting data.

Fire PSA utilizes the results of *deterministic fire analysis*. Difficult issues for the latter are e.g. what is the reason and the probability of the fire event, how the fire will develop etc. Definition of the magnitude of the fire as the function of the time is very difficult. A big problem is also the shortage of available statistical data. There is urgent need for organizing international co-operation in gathering appropriate data. Another developmental need is for tailored research for the Finnish plants.

Human reliability analysis (HRA) is usually carried out as a part of the PSA. The starting point of the PSA in the Finnish nuclear power plants is that automation has to take care of the disturbances and that there must be enough redundancy. The aim is to increase structural fire safety in order to reduce the human effect to the minimum. According to the interviewees, human actions are considered essential in the fire risk analysis but the degree they have been included in the model is considerably low. The modeling has been adopted from the analyses concerning internal events, which means that the special features of the fire situations have not been taken into account. The effects of the recovery actions made by operators have not been included in fire HRA. In the analyses, the operators do not intervene although in reality the situation could be improved by them. This is a conscious choice because there is not enough evidence to show that a certain, so far unmodeled human action would be important in fire situations. In addition, there are no approved means for analyzing the loss of information from the human point of view. When compared to the internal events, an evaluation of human intervention in fire situations would require paying attention to e.g. problems caused by increased stress. There is no clear logic according to which the probability parameters could be modified. Taking account of the human aspect could be enhanced but it would require more detailed procedures which is, however, problematic, e.g. due to the fact that cable routes are not known completely. Apart from procedural changes the increase of human reliability could also be achieved by educating attitudes towards occupational safety and health and organizational safety culture. However, most HRA methods are insensitive to such parameters.

Discussion

On the basis of the results of the expert interviews it seems that he inherent uncertainty, vagueness and complexity of NPP fire situations make them special when compared to process disturbances. Fires may reduce the available time, resources of decision making and operation and physical and psychic capacity of the persons involved in the control of fire situations. The demands of decision making are high and sometimes conflicting.

There are, however, ways to reduce these problems. Organizational support is needed for creating adequate prerequisites of decision making in fire situations. It would be desirable that experts representing different perspectives to fire situations would co-operate for the fulfillment of the developmental needs. It would also be fruitful to regard the different actors involved in the control of fire situations as a co-operational whole. The actors’ needs concerning procedures and training should be considered against the demands relating to the roles of these parts. An important issue would be creation of a link between the results of the PSA and operational activity in the control room, in order to widen the perspective of the operators. Also the relevant results of the deterministic fire analysis should be communicated to the operators. Furthermore, the practical ways to increase the communal fire fighters' knowledge of the plant and the process and to enhance the informativeness of communication between the control room and the fire brigade could be discussed. These kind of policies would enhance the professional skills of the different parts and probably lead to increase of interest to own work, too. They would also contribute to diminishing of possible stress and fear, which is of importance. The psychological effect of the potential fires may be comprehensive although the risk of them may be low.

In the next phase of the study the results will be utilized as a basis for the development of a more detailed task analysis concerning the demands of operators' decision making in fire situations. The analysis is aimed to support human reliability analysis and to serve as a conceptual tool for the development of procedures and training. The tool is intended to enhance mutual understanding between different experts related to fire situations.

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6. Summary

The main motif for public nuclear safety research is to guarantee independent resources for the Finnish regulatory body in the safety evaluation of nuclear power production. This includes development of tools and practices, education of experts and efficient communication about the latest technology and know-how. The results of the public research are also at the disposal of the nuclear power companies. During the last 10 years various fields of nuclear safety research have been organised as national research programmes. They have been launched and administrated by the Ministry of Trade and Industry. In the current Finnish Research Programme on Nuclear Power Plant Safety FINNUS (1999 – 2002) the fields of structural integrity and operational safety have been combined into three main themes of ageing, accidents and risks. A total of 11 research projects are being conducted under these themes. The annual volume is about FIM 22 million (€ 3.6 million) and 30 person years. The programme is conducted mainly at the Technical Research Centre of Finland (VTT) and the Lappeenranta University of Technology and coordinated by VTT Energy.

In the steering and selection of research goals of the FINNUS programme, the Radiation and Nuclear Safety Authority STUK has a dominant role (Appendix D). The research is also mainly funded from public sources. In order to combine limited national resources, the expertise of the power companies is also exploited in the steering and reference groups of the programmes. The power companies also contribute to funding of selected research topics of the FINNUS programme. In parallel with the national programme, other technology based programmes are under way, where the research goals have mainly been defined from the utility point of view.

Advances in the ageing field

The effects of ageing on nuclear power plants (NPP) are studied intensively, in order to evaluate the safe remaining lifetime of the components and the efficiency of possible corrective measures. In the FINNUS programme the field is covered with studies in material sciences of metallic structures, structural integrity studies and in-service inspection and monitoring methods.

In the material studies, new autoclave testing methods with multiple test cells, to simulate real loading conditions of NPP components, were devised and tested successfully to understand environmentally assisted cracking phenomena. A fairly good understanding of oxide film behaviour on the component surfaces was obtained by modelling transport of radioactive species through the film. In parallel, experimental techniques called controlled-distance electrochemistry was developed to measure properties of the oxide layer. It was also shown that the controlled-distance electrochemistry arrangement in solid contact can be used for the definition of fuel cladding corrosion. Studies on the effects of irradiation, annealing and reirradiation of pressure vessel material embrittlement for the Loviisa plant are going on. They include both theoretical studies and characterising the use of sub-size and reconstructed specimen for fracture toughness tests.

In the structural integrity field the ultimate goal is to develop a computational methodology to estimate safety margins in respect to failure mechanisms during the service life of NPP components. Such an approach includes detailed material characterisation. Particularly fracture toughness for brittle fracture was investigated by testing and modelling. The experiments indicated that, contrary to fracture mechanics theory, miniature size specimens underestimated the large specimen fracture toughness. The application range of the Master Curve approach for fracture toughness is being extended to low constraint geometries. Micromechanical computational methods are developed for the assessment of ductile fracture resistance. In the analysis of thermal and mechanical loads on structures, computational fluid dynamics calculations were performed when preparing to a test, that simulates effects of fluid thermal stratification on the piping structure. In another case, hydrogen detonation load on a reinforced concrete wall was simulated with specific detonation codes. The load data were transferred over to a structural analysis code, and corresponding structural calculations were carried out.

Planning and preparation of in-service inspection and monitoring was facilitated by acquiring two computer codes for simulation of ultrasonic testing. In the development of the Finnish qualification scheme for nondestructive testing, methods were developed for production of useful artificial defects, that may be applied in validation trials. Structural details of the Finnish nuclear power plants were surveyed in search for applicable measuring techniques, that are needed for condition monitoring of reinforced concrete structures.

Advances in the accident field

The accident theme in the FINNUS programme covers operational aspects of nuclear power plant safety. Research is conducted in the fields of nuclear fuel, reactor physics and dynamics, thermal-hydraulics and severe accidents.

The behaviour of high burnup fuel in accidents needs to be known in more detail and confidence. Extensive international programmes are being brought about to this end. Contributions and participation in the programmes are actuated in the FINNUS programme, with the note that the part covering the participation in the newly launched CABRI Water Loop Project on reactivity initiated accidents, will continue under a new management. Technical contributions include implementing fuel and cladding creep models in a French fuel transient code and an improved thermal hydraulics model in an

American code, the latter work only nearing completion. Applications of the stationary modelling of fuel were extended in a number of ways to cope with the fuel rod specific features and demands of higher burnup. As an example, because of the very localised build-up of plutonium, the peripheral layer of the fuel needed a more sophisticated description.

In reactor physics, the Monte Carlo method was increasingly applied for the solution of complex problems. It was also demonstrated that use of a three-dimensional transport code considerably improved accuracy (over 2D) of neutron and gamma dose estimates in reactor components. Another type three-dimensional neutronics model was developed, that is applicable in nodal reactor analysis codes. It enables more accurate calculation of new nuclear fuel types and reflector areas, which also improves out-of-core detector flux estimates. At the same time, a separate model was developed to improve out-of-core detector signal interpretation of the VVER reactors. The three-dimensional reactor dynamics code for square lattices TRAB-3D, developed at VTT, was tested and validated using both PWR and BWR circuits. All phases of the OECD/NEA main steam line break benchmark (MSLB TMI-1) were calculated successfully with the code. More detailed fuel models were worked out for the reactor dynamics codes. A new numerical scheme was generated for the advanced solution method of thermal-hydraulics equations, developed at VTT, and testing range of the method was extended.

New experiments were conducted with the PACTEL thermal-hydraulic test facility, that models the Loviisa power plant. Strong flow and power oscillations were recoded in an ATWS (Anticipated Transient Without Scram), where reactivity feedback of power from coolant density was included. The effect of collector header rupture size in a steam generator was tested applying the current operator emergency rules at Loviisa. The effect of non-condensable gases on horizontal steam generator performance was tested. The test loop for experiments on thermal stratification in a T-joint was planned and the first tests are being performed. In the thermal-hydraulic code validation excercises, the first phase of the MSLB TMI-1 benchmark was also calculated with the APROS code, developed by VTT and Fortum. The PACTEL loop model of the code was improved, as well as condensation description in the code. The collector header rupture experiments were simulated with an international code.

In severe accident management an essential goal is to evaluate pressure vessel failure mode and time in different accident scenarios. For this purpose VTT has developed the PASULA/FEM code. In the current programme the code was validated against the Sandia laboratory pressure vessel creep rupture tests at high primary pressure. VTT conducted in-house experiments on heat transfer in dry particle beds at high temperatures, which validated the PASULA heat transfer model for those conditions. A new test series on wet particle bed dryout is being designed. A novel threat of hydrogen accumulation in the Olkiluoto reactor building during a severe accident has been identified, because detonation of such a cloud could damage containment penetrations. The case was analysed by calculating formation of the hydrogen/air cloud with a computational fluid dynamics code and the detonation was calculated with two special codes. The first load data on structures were transferred to a structural analysis code, mentioned above. In the evaluation of organic iodine behaviour during a severe accident, different methods to reduce source term of methyl iodine were sought out.

Advances in the risk field

In the field of risk studies, one of the goals has been to concentrate on advanced risk analysis methods and their applicability, and on the other hand, to pay attention on risk or reliability evaluation of a certain process or technology. In the FINNUS programme the latter group includes studies on fire risks, safety critical applications of software based technology, as well as human and organisational performance.

In the introduction of advanced risk analysis methods, an expert panel methodology was developed for risk informed decision making. The concept was successfully tested by STUK in a risk informed in-service inspection pilot study, where expert panel was used to combine the deterministic information on degradation mechanisms and probabilistic information on pipe break consequences. Uncertainty assessment, that is a normal part of probabilistic safety assessment in the form of simulation of parametric uncertainties, was extended to physical processes and model uncertainty. As a special application, passive system reliability and licensing requirements were studied. Various risk importance measures were also surveyed and reported. Finally, development of human reliability analysis methods for fire risk assessment needs is currently ongoing as a co-operation between risk analysis specialists and psychologists.

In fire safety research, a universal calculation method for the critical heating time of control equipment at NPPs was proposed for equipment contained in metal boxes. Acute effects of smoke on insulation resistance of electronics used in programmable automation circuitry in NPPs were studied experimentally and theoretically. Quantitative models for soot accumulation, loss of insulation resistance and dimensioning of insulation coating were proposed. For real commercial circuits, coated by a protective lacquer layer, no electrical changes were observed. Data collection and reduction work for extracting reliability data of fire detection and sprinkler systems in both nuclear and non-nuclear installations was started. Probabilistic methods in fire risk assessment were demonstrated in a tunnel scenario, where a cable shelf fire may ignite the redundant cabling on the opposite wall of the tunnel. A fire simulation code, in combination with a commercial risk analysis package, was used in the evaluation of ignition probability.

In the reliability evaluation of safety critical software based automation systems, the focus is on the quantitative reliability analysis, with the aim to support the authorities and utilities in the licensing issues. In the development of reliability assessment methods, the Bayes networks was selected as the basic tool, because it allows transparent combination of various types of evidence needed in the reliability evaluation. In order to test applicability of the developed Bayes network model, suitable case studies are being planned, the first one probably addressing software controlled relays. Ageing of instrumentation and control equipment was surveyed in a special study. The major finding was that, rather than physical ageing of the components, technological obsolescense and ever shorter life span of the replacing products are more important problems.

In the research on working practices and safety culture, the core task analysis was developed further with a focus on current and future practices of process control work. The human reliability analysis in the fire scenario, mentioned above, concentrated on conceptualisation of cognitive demands in process control. In the organisational and safety culture research, the basic concepts were specified and a methodology was proposed to address safety culture within high reliability organisations. In this connection, a survey was conducted on organisational culture and identification of core mission and requirements of the Nuclear Reactor Regulation department of STUK. In the analysis of coping with demanding situations, a motivational performance model was developed. In this model, expert identity in relation to the surrounding community is a central concept, that is composed of meaningfulness of a person’s work, professional self-confidence and sense of control.

Statistics

During the period covered in this report, the FINNUS programme produced a total of 263 reports in various categories. A major publishing form has been conference presentations. A lot of detailed technical results were documented as working reports with limited distribution (Appendix A).

International co-operation was intensive in all the research fields. In addition to publishing activities in various international fora, the research staff contributed to working groups and networks, as well as defined and solved international benchmarks and participated in round robin excercises (Appendix B).

The research programme contributed to education of new experts in the field of nuclear safety in co-operation with the universities. Three doctoral theses, one licentiate and eight master’s theses were compleded (Appendix C). A number of under graduate students participated in the research projects as summer trainees or with a purpose to graduate in the future.

Appendix A: Publications of the projects in 1999–30.8.2000

Table. Publications of the FINNUS projects in 1999–30.8.2000.

Project	Scientific publications	Conf. papers	Research institute reports	Others	**Total**
AGE	7	21	1	4	**33**
STIN	1	9	-	6	**16**
INSMO	-	-	-	5	**5**
KOTO	-	4	-	14	**18**
READY	-	18	2	32	**52**
TOKE	1	6	-	11	**18**
MOSES	-	5	6	25	**36**
FISRE	-	9	5	6	**20**
PASSI	-	-	-	3	**3**
METRI	6	12	7	7	**32**
WOPS	2	10	5	6	**23**
HALTI	-	4	-	3	**7**
Total	**17**	**98**	**26**	**122**	**263**

Ageing Phenomena (AGE): publications 1999–30.8.2000

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Vanttola, T. FINNUS national research programme on nuclear safety. 3^{rd} Finnish-French colloquium on nuclear power plant safety. 27.–28.6.2000, Lappeenranta, Finland.

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Vanttola, T. Researching nuclear reactor safety. Energy in Finland 1999.

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Appendix B: International co-operation of the projects in 2000

OECD/Nuclear Energy Agency

Committee on the Safety of Nuclear Installations (CSNI) was restructured at the beginning of 2000. The current representatives are

CSNI, L. Mattila, VTT Energy

* Working Group on Risk Assessment, P. Pyy, VTT Automation

* Working Group on Accidents and Management of Accidents, R. Sairanen, VTT Energy

* Working Group on Integrity and Ageing of Components and Structures, R. Rintamaa, VTT Manufacturing Technology

  Task Group on Integrity of Metal Components and Structures, J. Solin, VTT Manufacturing Technology

* Working Group on Fuel Safety Margins, K. Ranta-Puska, VTT Energy

* CSNI Programme Review Group, T. Vanttola, VTT Energy

* Special Expert Group on Human and Organisatorial Factors, L. Norros, VTT Automation

* Special Expert Group on Fuel Safety Margins, K. Ranta-Puska, VTT Energy

Nuclear Science Committee (NEANSC), M. Anttila, A. Tanskanen, VTT Energy

* *Task force on scientific issues of fuel behaviour*, S. Kelppe, VTT Energy

Halden Reactor Project/Halden Programme Group, P.Pyy (Vice chairman), VTT Automation, K. Ranta-Puska, VTT Energy

* *Advanced Monitoring techniques for Application in Material Studies,* K. Mäkelä, VTT Manufacturing Technology
* *Pressure Vessel Ageing,* M. Valo, VTT Manufacturing Technology

* *Irradiation Assisted Stress Corrosion Cracking*, P. Aaltonen, VTT Manufacturing Technology
* *Fuel performance analysis,* K. Ranta-Puska, VTT Energy
* *Human reliability*, J. Kettunen, P. Pyy, VTT Automation
* *Reliability of software based control systems,* U. Pulkkinen, P. Haapanen, VTT Automation
* *Maintenance decision making*, K.Laakso, VTT Automation

International Atomic Energy Agency

Co-ordinated Research Projects

Round-robin Exercise on WWER-440 RPV Weld Metal Irradiation Embrittlement, Annealing, and Re-embrittlement, M. Valo, VTT Manufacturing, Technology

International Working Group of Life Management of Nuclear Power Plants (IWG-LMNPP), K. Wallin, VTT Manufacturing Technology

High temperature On-line Monitoring of Water Chemistry and Corrosion, (WACO), K. Mäkelä, VTT Manufacturing Technology

CRP Coordinated Research Programme "Assuring Structural Integrity of Reactor Pressure Vessels", K. Wallin, VTT Manufacturing Technology

International Working Group on Water Reactor Performance and Technology (IWGFPT), R. Teräsvirta, Fortum Engineering, S. Kelppe, VTT Energy

International Working Group on Nuclear Power Plant Control and Instrumentation (NPPCI), B. Wahlström, VTT Automation

Development of plant-specific safety indicators, E. Lehtinen, VTT Automation

Applicability of Computer Code for Safety Analysis of New Fuels for WWER Reactors, K. Ranta-Puska, VTT Energy

Commission of the European Communities

DGXVII, European Forums

* *European Plant Life and Ageing Forum (EPLAF)* R. Rintamaa, VTT Manufacturing Technology
* *European NDE Forum (ENDEF),* Pentti Kauppinen, VTT Manufacturing Technology

DGXI, *Working group codes and standards (WGCS),* R. Rintamaa

DGXII Networks coordinated by JRC/IAM, R. Rintamaa

* *Network for Evaluating Steel Components (NESC),* AG1 Inspection, Pentti Kauppinen and Jorma Pitkänen, VTT Manufacturing Technology
* *European Action Group for RPV Ageing Materials Evaluation and Studies (AMES)*
* *European Network for Inspection Qualification (ENIQ),* Pentti Kauppinen, VTT Manufacturing Technology

Nuclear Fission Safety in the Fifth Framework Programme

* *External Advisory Group (EAG),* L. Mattila, VTT Energy
* *Consultative Committee Euratom-Fission (CCE-Fission),* T. Haapalehto, KTM
* *Evaluation of non destructive testing techniques for monitoring of material degradation (GRETE*), P. Kauppinen, VTT Manufacturing Technology
* *High Performance Light Water Reactors*, R. Kyrki-Rajamäki, VTT Energy

Cooperation on VVER Reactor Physics and Dynamics (AER), Scientific council, R. Kyrki-Rajamäki, VTT Energy

* *Working group on VVER Reactor Safety Analysis*, R. Kyrki-Rajamäki, VTT Energy
* *Working group on physical problems of spent fuel, radioactive waste and decommissioning of nuclear power plants,* A. Tanskanen, VTT Energy
* *AER annual symposiums*, N.N., VTT Energy

Nordic Nuclear Safety Research (NKS), Steering group, L. Mattila, VTT Energy

* *Reactor safety, NKS/SOS-2 project,* K.Simola, U. Pulkkinen, K. Laakso, P.Pyy, VTT Automation

* *NKS/ SOS-2 Project: Reactor Safety 1998-200, Subproject SOS.2.3: Severe accidents, Subtask 3: Hydrogen issues,* A. Silde, VTT Energy
* *NKS/SOS-1 project,* B. Wahlström, VTT Automation

Scientific Communities

European Safety, Reliability and Data Association (ESReDA)

* *General Secretary, Organisation of yearly ESReDA Seminars,* P. Pyy, VTT Automation

European Safety and Reliability Conferences (ESREL), P. Pyy, VTT Automation

* Yearly oparticipation in organising committee

European Structural Integrity Society (ESIS), K. Wallin, H. Talja, VTT Manufacturing Technology

* Development of European "standards" on fracture mechanics, information exhance. VTT co-chairs the Materials Task Group and articipates in the Numerical task Group

International Group for Radiation Damage Mechanisms in Pressure Vessel Steels (IGRDM), K. Wallin, M. Valo, VTT Manufacturing Technology

ASTM, K. Wallin, E-10, M. Valo, VTT Manufacturing Technology

* *subcommittee E-10 on Nuclear Technology* concentrates on monitoring of irradiation embrittlement using small specimens and developes related standards.

ASME, R. Rintamaa, VTT Manufacturing Technology

* participation in yearly PVP (Pressure Vessels and Piping) conferenses

Cooperation with various institutes

SCK*CEN, Studiecentrum voor Kernenergie, Belgium

* *Development of advanced monitoring techniques,* K. Mäkelä, VTT Manufacturing Technology

Paul Sherrer Institut, Switzerland

*	European Round Robin on Constant Load EAC Tests of Low Alloy Steel under BWR Conditions, P. Karjalainen-Roikonen, U. Ehrnstén, VTT Manufacturing Technology

Leningrad NPP Sosnovyj Bor, Russia

- *The Finnish-Russian co-operation on integrity of pressurised components,* Pentti Kauppinen, VTT Manufacturing

Institute de Protection et de Surete Nucleaire, Cadarache, France

Study of the behaviour of highly irradiated fuels in case of reactivity accident and the SCANAIR computer code, S. Kelppe, VTT Energy

- *OECD-IPSN CABRI Water Loop Project 2000-2007. Umbrella Agreement with OECD, bilateral Agreement with IPSN; jointly with Fortum Power and Heat Oy and Teollisuuden Voima Oy*. S Kelppe VTT Energy.

Research Institute of Technology, NITI, Russia

- *Scientific cooperation on thermal-hydraulic experiments,* H. Purhonen, VTT Energy

US Nuclear Regulatory Commission (USNRC)

- *Code Application and Maintenance (CAMP),* H. Holmström, VTT Energy
- *Co-operative Severe Accident Research Programme (CSARP),* R. Sairanen, VTT Energy
- *FRAPCON-3 Code Users' Group,* S. Kelppe, VTT Energy
- *Co-operative PRA search Programme (COOPRA),* P.Pyy, VTT Automation (observer)

Electric Power Research Institute (EPRI)

- *Advanced Containment Experiments, Extension (ACEX),* I. Lindholm, VTT Energy
- *Melt Attack and Coolability (MACE),* I. Lindholm, VTT Energy

Institute de Protection et de Surete Nucleaire, Cadarache, France

- *Human reliabilitys,* P.Pyy, VTT Automation

Swedish Nuclear Power Inspectorate (SKI), Sydkraft and Vattenfall Ab, Sweden

- Forskningnämnd, L. Norros, VTT Automation
- *Statistical methods, decision analysis, human errors, maintenance and PSA,* K. Laakso, P. Pyy, VTT Automation

JRC Ispra, Italy

* *Expert judgement*, U. Pulkkinen, VTT Automation

Staatliche Materialprüfungsanstalt (MPA), Germany
* *materials research*, R. Rintamaa, VTT Manufacturing Technology

Fraunhofer-Institut für Werkstoffmechanik (IWM), Germany
* *structural analysis and computational fracture mechanics, especially development of new material models*, H. Talja, VTT Manufacturing Technology

University of Illinois, USA
* *computational fracture mechanics, assessment of damage,* K. Wallin, VTT Manufacturing Technology

Forschungszentrum Karlsruhe (FZK) , Germany
* *Acquisition and User Training of 3D Hydrogen Detonation Computer Code, DET3D*, Ari Silde

Other co-operation

International Co-operative Group on Environmentally Assisted Cracking of Light Water Reactor Materials (ICG-EAC), P.Aaltonen, VTT Manufacturing Technology

International Group on Radiation Damage Mechanisms in Pressure Vessel Steels (IGRDM), M.Valo and K.Wallin, VTT Manufacturing Technology

Nordic Reactor Physics Meetings "Reactor Physics Calculations in the Nordic Countries", R. Höglund, VTT Energy

MOSAIC group (international group of developers of human reliability analysis methods), P.Pyy

PSAM – probabilistic safety assessment and management conferences, P. Pyy, VTT Automation

European Association of Cognitive Ergonomics (EACE), L. Norros, VTT Automation

New Technology and Work (NeTWork), L. Norros, VTT Automation

Work process knowledge in technological and organizational development (WHOLE), Thematic network, TSER, L. Norros VTT Automation

International Collaborative Project to Evaluate Fire Models for Nuclear Power Plant Applications organized by US Nuclear Regulatory Commission (USNRC), O. Keski-Rahkonen, VTT Building Technology

Appendix C: Academic degrees awarded

Academic degrees in AGE project in 1999–30.8.2000

Doctor of Philosophy:

Mäkelä, Kari. Development of techniques for electrochemical studies in power plant environments, Espoo 2000: Technical Research Centre of Finland, VTT Publications 415, 46 pp. + app. 128 pp. (University of Helsinki.)

Academic degrees in KOTO project in 1999–30.8.2000

Master of Science in Technology:

Knuutila, Arttu. Creep model for a nuclear fuel rod. Espoo, 64 p. (Helsinki University of Technology.)

Academic degrees in READY project in 1999–30.8.2000

Master of Science in Technology:

Mattila, Riku. Pin Power Reconstruction Module for the TRAB-3D Nodal Code. Espoo, 58 p. (Helsinki University of Technology.)

Latokartano, Saku. Kiehutusvesireaktorin lataussuunnitteluohjelman CORFU:n kehittäminen (Development of the BWR reload planning program CORFU). Espoo, 81 p. (in Finnish) (Helsinki University of Technology.)

Academic degrees in TOKE project in 1999–30.8.2000

Doctor of Technology:

Puska, Eija Karita. Nuclear reactor core modelling in multifunctional simulators. Espoo 1999: Technical Research Centre of Finland, VTT Publications 376. 67 p. + app. 73 p. (Helsinki University of Technology.)

Licentiate of Technology:

Miettinen Jaakko, Thermohydraulic Model SMABRE for Light Water Reactor Simulations, Espoo, 151 p. (Helsinki University of Technology.)

Master of Science in Technology:

Leppänen Ari-Pekka., Ydinvoimalaitoksen primääripiiriä kuvaavan PACTEL-koelaitteiston mallinnus ja kelpoistus APROS 5 simulointiympäristössä. Lappeenranta, 84 p. (In Finnish) (Lappeenranta University of Technology).

Tolonen, Pekka. User's guide for advanced process simulation software APROS 5. Lappeenranta, 75 p. + app. 5 p. (Lappeenranta University of Technology.)

Academic degrees in MOSES project in 1999–30.8.2000

Master of Science:

Karhu, Anna. Methods to Prevent the Source Term of Methyl Iodide during a Core Melt Accident, 1999, 59 p. (University of Jyväskylä).

Academic degrees in PASSI project in 1999–30.8.2000

Master of Science in Technology:

Helminen, Atte. Reliability estimation of software-based digital systems using Bayesian networks. Espoo, 50 p. (Helsinki University of Technology.)

Academic degrees in METRI project in 1999–30.8.2000

Doctor of Technology:

Simola, Kaisa. Reliability methods in nuclear power plant ageing management. VTT Publications 379, 1999. 38 p. + app. 96 p. (Helsinki University of Technology.)

Academic degrees in WOPS project in 1999–30.8.2000

Master of Arts in Psychology:

Reiman, T. 2000: Organisaatiokulttuuri Säteilyturvakeskuksen Ydinvoimalaitosten valvonta -osastolla. 79 s. (In Finnish). (University of Helsinki).

Appendix D: The steering group, the reference groups and the scientific staff of the projects and their tasks in 2000

THE STEERING GROUP OF THE FINNISH RESEARCH PROGRAMME ON NUCLEAR POWER PLANT SAFETY (FINNUS), 1999–2002

YDINVOIMALAITOSTEN TURVALLISUUSTUTKIMUSOHJELMAN (1999–2002) JOHTORYHMÄ

Person	Organisation	
Lasse Reiman	STUK	Chairperson
Heikki Kalli	LTKK	Member
Seija Suksi	STUK	Member
Rauno Rintamaa	VTT VAL	Member
Riitta Kyrki-Rajamäki	VTT ENE	Member
Petra Lundström	FORTUM ENG.	Member
Ralf Ahlstrand	FORTUM ENG.	Member
Marjo Mustonen	TVO	Member
Eero Patrakka	TVO	Member
Reijo Munther	TEKES	Expert
Timo Haapalehto	KTM	Expert
Anne Väätäinen	KTM	Expert
Urho Pulkkinen	VTT AUT	Expert

RERFERENCE GROUPS OF THE FINNISH RESEARCH PROGRAMME ON NUCLEAR POWER PLANT SAFETY (FINNUS), 1999–2002

YDINVOIMALAITOSTEN TURVALLISUUSTUTKIMUSOHJELMAN (1999–2002) PROJEKTIEN TUKIRYHMÄT

Ageing phenomena (AGE)
Laitosten ikääntymisilmiöt

Person	Organisation
Rainer Rantala (deputy chairperson)	STUK
Juhani Hinttala	STUK
Hannu Hänninen (chairperson, on leave)	HTKK
Ossi Hietanen	FORTUM ENG.
Jyrki Kohopää	FORTUM ENG.
Thomas Buddas	FORTUM Loviisa
Asko Alho	FORTUM Loviisa
Erkki Muttilainen	TVO
Anneli Reinvall	TVO
Risto Sairanen	VTT ENE
Kaisa Simola	VTT AUT

Structural Integrity (STIN)
Lujuuden varmistaminen

Person	Organisation
Rauli Keskinen (chairperson)	STUK
Rainer Rantala	STUK
Mauri Määttänen	TKK
Jyrki Kohopää	FORTUM ENG.
Alpo Neuvonen	FORTUM ENG.
Paulus Smeekes	TVO
Heikki Purhonen	VTT ENE
Heli Talja	VTT VAL

In-service inspections and monitoring (INSMO)
Tarkastukset ja kunnonvalvonta

Person	Organisation
Juhani Hinttala (chairperson)	STUK
Olavi Valkeajärvi	STUK
Hannu Hänninen	TKK
Kari Hukkanen	TVO
Raimo Paussu	FORTUM ENG
Kalervo Orantie	VTT RTE

Behaviour of high burnup fuel in accidents (KOTO)
Korkeapalamaisen polttoaineen transientti- ja onnettomuuskäyttäytyminen

Reactor physics and dynamics (READY)
Reaktorifysiikka ja dynamiikka

Person	Organisation
Keijo Valtonen (chairperson)	STUK
Matti Ojanen	STUK
Martti Antila	FORTUM ENG.
Pertti Siltanen	FORTUM ENG.
Risto Teräsvirta	FORTUM ENG.
Seppo Koski	TVO
Esa Mannola	TVO
Ralf Lunabba	TVO
Risto Sairanen	VTT ENE
Lena Hansson-Lyyra	VTT VAL

Thermal hydraulic experiments and code validation (TOKE)
Termohydrauliset kokeet ja ohjelmistojen kelpoistus

Person	Organisation
Juhani Hyvärinen (chairperson)	STUK
Hannu Ollikkala	STUK
Harri Tuomisto	FORTUM ENG.
Samuli Savolainen	FORTUM Loviisa
Heikki Sjövall	TVO
Juhani Vihavainen	LTKK
Markku Rajamäki	VTT ENE
Markku Hänninen	VTT ENE
Arja Saarenheimo	VTT VAL

Modelling and simulant experiments of severe accident phenomena (MOSES)
Vakavien reaktorionnettomuuksien mallinnus ja simulanttikokeet

Person	Organisation
Kalevi Haule (chairperson)	STUK
Juhani Hyvärinen	STUK
Harri Tuomisto	FORTUM ENG.
Olli Kymäläinen	FORTUM ENG.
Heikki Sjövall	TVO
Seppo Koski	TVO
Pertti Auerkari	VTT VAL
Kaisa Simola	VTT AUT
Markku Anttila	VTT ENE

Fire safety research (FISRE)
Paloturvallisuustutkimus

Person	Organisation
Reino Virolainen (chairperson)	STUK
Jouko Marttila	STUK
Antti Norta	FORTUM ENG.
Kari Taivainen	TVO
Mika Yli-Kauhaluoma	TVO
Pekka Kallioniemi	Teollisuusvakuutus Oy
Pekka Pyy	VTT AUT
Mikko Manninen	VTT ENE
Juho Saarimaa	VTT RTE

Programmable automation system safety integrity assessment (PASSI)
Ohjelmoitavan automaation turvallisuuden arviointi

Person	Organisation
Marja-Leena Järvinen (chairperson)	STUK
Reino Virolainen	STUK
Samuel Koivula	STUK
Arto Felin	FORTUM ENG.
Markku Winter	FORTUM ENG.
Risto Himanen	TVO
Jukka Pulkkinen	TVO
Urho Pulkkinen	VTT AUT
Olli Ventä	VTT AUT
Björn Wahlström	VTT AUT
Timo Okkonen	VTT ENE

Methods for risk analysis (METRI)
Riskianalyysin menetelmät

Person	Organisation
Reino Virolainen (chairperson)	STUK
Ilkka Niemelä	STUK
Jussi Vaurio	FORTUM Loviisa
Kalle Jänkälä	FORTUM ENG.
Risto Himanen	TVO
Olavi Keski-Rahkonen	VTT RTE
Ilona Lindholm	VTT ENE
Heli Talja	VTT VAL
Olli Ventä	VTT AUT

Working practices and safety culture in nuclear power plant operations (WOPS)
Toimintatavat ja turvallisuuskulttuuri ydinvoimalaitoksessa

Person	Organisation
Anneli Leppänen (chairperson)	TTL
Petteri Tiippana	STUK
Kaisa Åstrand	STUK
Pekka Kettunen	FORTUM Loviisa
Matti Kattainen	FORTUM ENG.
Markku Malinen	TVO
Pekka Pyy	VTT AUT
Kaarin Ruuhilehto	VTT AUT
Lasse Nurmi	Poliisiammatti-korkeakoulu

Personnel of the Ageing phenomena (AGE) project in 2000

Person	Tasks
Pertti Aaltonen, MScTech	Project manager, Environmentally assisted cracking of nuclear materials
Päivi Karjalainen-Roikonen, MScTech	Materials' behaviour in nuclear environments, fracture mechanics
Aki Toivonen, MScTech	Irradiation assisted stress corrosion cracking, experimental fracture mechanics
Pekka Moilanen, MScTech	Experimental methods for environmentally assisted cracking
Ulla Ehrnstén, MScTech	Microscopy, materials' behaviour in nuclear environments
Gary Marquis, DTech	Fatigue testing and analysis methods
Timo Laitinen, DTech	Electrochemistry, water chemistry, corrosion phenomena in nuclear power plants
Kari Mäkelä, PhD	Water chemistry, electrochemistry, corrosion phenomena in nuclear power plants, activity build-up
Timo Saario, DTech	Surface film measurement techniques
Martin Bojinov, DTech	Development of oxide film models
Pertti Koukkari, DTech	Modelling of porous oxide films
Markus Olin, PhD	Modelling of porous oxide films, surface complexation
Pekka Sten, MSc	Surface complexation
Jarmo Lehikoinen, MSc	Modelling of porous oxide films
Matti Valo, MScTech	Neutron flux effects, re-embrittlement and repair methods
Kim Wallin, Prof	Fracture mechanics, statistical analysis
Reijo Pelli, MScTech	NPP materials and repair methods
Lena Hansson-Lyyra, MScTech	Fuel cladding

Personnel of the Structural Integrity (STIN) project in 2000

Person	Tasks
Heikki Keinänen, MscTech	Project manager, structural analysis
Timo Pättikangas, DTech	Computational fluid dynamics
Anssi Laukkanen, MscTech	Structural analysis, fracture mechanics
Gary Marquis, DTech	Fatique analysis
Markku Nevalainen, DTech	Structural analysis, fracture mechanics
Pekka Nevasmaa, MScTech	Welding technology
Tapio Planman, MScTech	Fracture mechanics
Arja Saarenheimo, LicTech	Structural analysis, impact loads, reinforced concrete
Jussi Solin, MScTech	Fatique analysis
Heli Talja, DTech	Structural analysis, fracture mechanics
Kim Wallin, DTech, Prof.	Fracture mechanics, miniature specimens
Heikki Purhonen, LicTech	Large scale fluid dynamics testing

Personnel of the In-service inspections and monitoring (INSMO) project in 2000

Person	Tasks
Matti Sarkimo, LicTech	Project manager; modelling of ultrasonic testing and comparison with real testing results.
Pentti Kauppinen, DTech	Assessment of the material properties by ultrasonic method.
Jorma Pitkänen, LicTech	Modelling of ultrasonic testing and comparison with real testing results.
Pauli Särkiniemi, BScTech	Analysis of test results obtained from qualification test samples.
Harri Jeskanen, Technician	Testing and analysis of test results obtained from qualification test samples.
Tapani Packalen, BScTech	Inspection methods of reinforced concrete structures.
Petri Kuusinen, BScTech	Assessment of the material properties by ultrasonic method.
Kalervo Orantie, MScTech	Inspection methods of reinforced concrete structures.
Erkki Vesikari, MScTech	Inspection methods of reinforced concrete structures.

Personnel of the High burnup fuel in accidents (KOTO) project in 2000

Person	Tasks
Seppo Kelppe, MscTech	Project manager, RIA-analyses, SCANAIR-code development, CABRI participation
Kari Ranta-Puska, MscTech	ENIGMA-development, Halden cooperation
Jan-Olof Stengård, Assistant research scientist	FRAPCON3/FRAPTRAN verification, maintenance of programmes
Kari Ikonen, LicTech	Supervision of mechanics studies
Arttu Knuutila, MScTech	SCANAIR development

Personnel of the Reactor physics and dynamics (READY) project in 2000

Person	Tasks
Antti Daavittila, MScTech	Deputy project manager, Reactor Dynamics, validation of TRAB-3D, application of CFDPLIM
Aapo Tanskanen, MScTech	Deputy project manager, Reactor Physics, Monte Carlo methods, criticality safety
Markku Anttila, MScTech	Reactor Physics, OECD/NEA connections, NEANSC (Science Committee)
Milja Eskola, MSc	Application of PLIM in the reactor dynamics codes
Anitta Hämäläinen, MScTech	Circuit modelling, dynamics benchmarks
Randolph Höglund, LicTech	Reactor Physics, Nordic connections
Elja Kaloinen, MScTech	Reactor Physics, validation of TRAB-3D
Riitta Kyrki-Rajamäki, DTech	Development and validation of three-dimensional dynamics codes with circuit models, International co-operation on VVER safety
Saku Latokartano, MScTech	Reactor Physics, code validation
Elina Lepistö, Student	Reactor Dynamics, fuel models in dynamics codes
Riku Mattila, MScTech	Reactor Physics, advanced nodal methods
Markku Rajamäki, DTech	Development, testing and application of CFDPLIM and SFAV
Hanna Räty, MScTech	Project manager, development and testing of applying PLIM in the reactor dynamics codes (improved thermal hydraulics modelling), validation of TRAB-3D Maternity leave November 1999 - November 2000
Timo Vanttola, DTech	Special questions on thermal hydraulics
Frej Wasastjerna, LicTech	Reactor Physics, MCNP (Monte Carlo-calculation)

The READY project also employs research trainees during the summer

Personnel of the Thermal hydraulic experiments and code validation (TOKE) project in 2000

Person	Tasks
Markku Puustinen, MScTech	Project manager, Experimental work , APROS analyses
Heikki Purhonen, LicTech	Analyses of experimental data
Vesa Riikonen, MScTech	Analyses of experimental data, computer system manager
Ismo Karppinen, MScTech	Thermal-hydraulic validation of the APROS code
Sixten Norrman, MScTech	Thermal-hydraulic validation of the APROS code
Eija Karita Puska, DTech	APROS analyses
Vesa Yrjölä, MScTech	Thermal hydraulics, RELAP5 analyses
Harri Partanen, Technician	PACTEL operation and maintenance
Hannu Pylkkö, Technician	PACTEL operation and maintenance
Ilkka Saure, Technician	PACTEL operation and maintenance, instrumentation and control systems

The project also employs part time and full time research trainees.

Personnel of the Modelling and simulant experiments of severe accident phenomena (MOSES) project in 2000

Person	Tasks
Ilona Lindholm, MScTech	Project manager, Experiments on debris coolability, MACE/ACEX follow-up
Risto Sairanen, LicTech	OECD/LHF follow-up
Kari Ikonen, LicTech	Development of the PASULA code, participation on OECD/Sandia programme
Jorma Jokiniemi, PhD	Fission product behaviour
Pertti Auerkari, MScTech	Debris coolability experiments
Stefan Holmström, MScTech	Debris coolability experiments
Esko Pekkarinen, MScTech	SA models for APROS code
Ari Silde, MScTech	Development of SA models for the APROS code, Hydrogen combustion studies
Ari Auvinen, MScTech	Fission product behaviour studies

Personnel of the Fire safety research (FISRE) project in 2000

Person	Tasks
Olavi Keski-Rahkonen, DTech	Project manager
Johan Mangs, LicPhil	Effect of smoke and heat on electronics
Simo Hostikka, MScTech	Utilisation of fire safety research for PSA
N.N.	Active fire protection equipment

There will be additional persons working on subcontract base from Helsinki University of Technology, as well as three students working on their master's theses.

Personnel of the Programmable automation system safety integrity assessment (PASSI) project in 2000

Person	**Task**
Urho Pulkkinen, DTech	Project manager, probability methods, reliability analysis, operational exåerience analysis
Kaisa Simola, DTech	Ageing analyses of I&C systems
Tony Rosqvist, LicTech	Ageing analysis of I&C systems
Atte Helminen, MScTech	Construction of Bayes network model, operational exåerience analysis, automation analysis
Pentti Haapanen, MScTech	Construction of Bayes network model, operational exåerience analysis, automation analysis
Teemu Tommila, MSc	Automation analysis
N.N.	Process & automation analysis

Personnel of the Methods for risk analysis (METRI) project in 2000

Person	Task
Pekka Pyy, LicTech	Project manager, international relations, human reliability assessment, PSA qualification, dynamic and integrated safety analysis, passive system reliability, human reliability
Kaisa Simola, DTech	Assistant project manager, NKS/SOS-2 project manager and research tasks, ageing, pipe break frequencies, decision panels
Urho Pulkkinen, DTech	Probability mathematics, uncertainty analysis, importance measures, dynamic methods, decision panels
Kari Laakso, DTech	Maintenance decision making (NKS task)
Tony Rosqvist, LicTech.	Decision analysis, uncertainty of physical models
Atte Helminen, MScTech	Passive system reliability
Eija Myötyri, Student	Masters thesis on importance measure assessment
N.N.	Decision analysis and uncertainties

Personnel of the Working practices and safety culture in nuclear power plant operations (WOPS) project in 2000

Person	Tasks
Maaria Nuutinen, MA	Deputy project manager, decision making in dynamic operational situations, analysis of the operators' actions in coping with demanding situations (PhD work)
Leena Norros, PhD	Project manager, decision making in dynamic operational situations, safety culture, human factors in NDT (on leave)
Kristiina Hukki, MA	Decision making in dynamic operational situations, development of the taxonomy of process situations (PhD work)
Jari Kettunen, MA	Analysis of safety indicators and quality control systems
Björn Wahlström, Research Prof.	Analysis of safety indicators and quality control systems
Teemu Reiman, MA	Analysis of safety culture and working practices
Pia Oedewald, MA	Analysis of working practices

Personnel of the Administration and information of the research programme (HALTI) project in 2000

Person	Tasks
Timo Vanttola, DTech	Programme leader
Eija Karita Puska, DTech	Project coordinator

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FINNUS
The Finnish Research Programme on Nuclear Power Plant Safety
Interim Report 1999 – August 2002

Abstract

FINNUS (1999–2002) is the Finnish public research programme on nuclear power plant safety, launched and administrated by the Ministry of Trade and Industry (KTM). The programme concentrates on the themes of ageing, accidents and risks. The general objectives of the programme are to develop tools and practices for safety authorities and utilities, to provide a basis for safety-related decisions, to educate new nuclear energy experts and to promote technology and information transfer. The technical objectives of the programme are prepared in the guidance of the Radiation and Nuclear Safety Authority (STUK), but the views of the power companies are taken into consideration. Funding of the programme is mainly from public sources. The annual volume of the programme is about FIM 22 million (€ 3.6 million) and 30 person years. The research is coordinated and mainly conducted by the Technical Research Centre of Finland (VTT).

The effects of ageing on nuclear power plants are studied intensively, in order to evaluate safe remaining lifetime of the components and efficiency of the corrective measures. The programme concentrates on studies in material sciences of metallic structures, structural integrity and in-service inspection and monitoring methods. The accident theme concerns operational aspects of nuclear power plant safety. The issues of nuclear fuel behaviour, reactor physics and dynamics modelling, thermal-hydraulics and severe accidents are addressed under the theme. In the risk field attention is paid on one hand on advanced risk analysis methods and their applicability, and on the other hand, on the evaluation of fire risks, safety critical applications of software based technology, as well as human and organisational performance.

The report summarises goals and results of the programme during the period 1999 – August 2000.

Keywords

FINNUS, nuclear power plants, reactor safety, materials, corrosion, ageing, accidents, thermal hydraulics, modelling, fire safety, risk analysis

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