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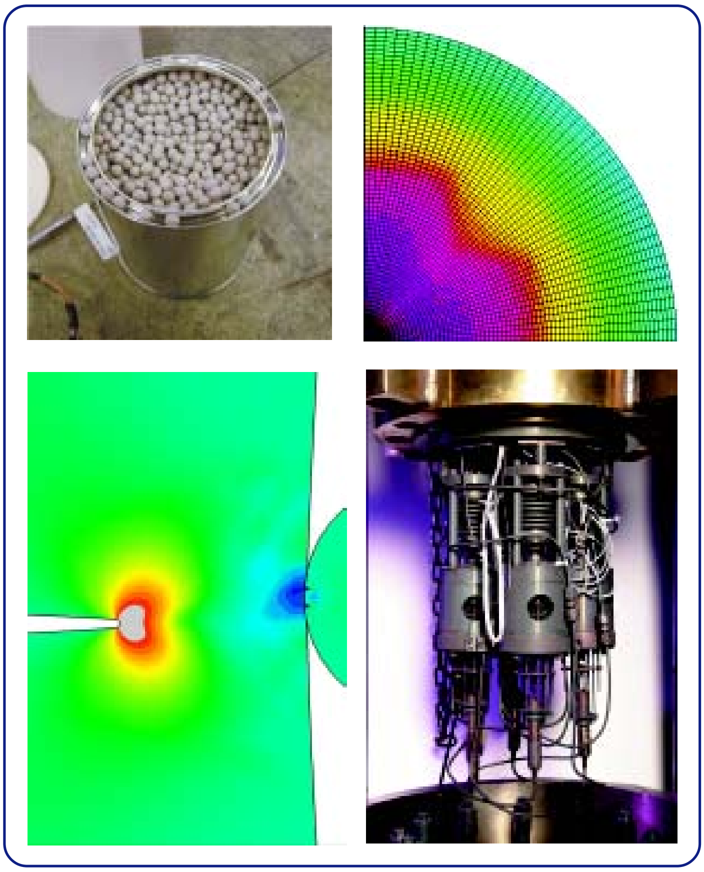

FINNUS
The Finnish Research Programme on Nuclear Power Plant Safety 1999–2002
Final Report

VTT TIEDOTTEITA – RESEARCH NOTES 2164

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Final Report

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VTT Processes

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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 has concentrated on the themes of ageing, accidents and risks. The general objectives of the programme have been 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 have been prepared under the guidance of the Radiation and Nuclear Safety Authority (STUK), but the views of the Finnish power companies have been taken into consideration. Funding of the programme has been mainly from public sources. The annual volume of the programme has been about Euro 3.6 million and 30 person-years. The research has been co-ordinated and mainly conducted by the Technical Research Centre of Finland (VTT) with a significant contribution from Lappeenranta University of Technology (LTKK).

The effects of **ageing** on nuclear power plants have been studied intensively in order to evaluate the safe remaining lifetime of the components and the efficiency of the corrective measures. The programme has mainly concentrated on studies in ageing effects on material properties and degradation mechanisms of metallic structures, structural integrity and in-service inspection as well as monitoring methods including reinforced concrete structures as a new area. The **accident** theme has concentrated on operational aspects of nuclear power plant safety. The issues of nuclear fuel behaviour, reactor physics and dynamics modelling, thermal hydraulics and severe accidents were addressed under the theme by conducting both computational and experimental studies. In the **risk** field, attention has been paid to advanced risk analysis methods and their applicability, and to the evaluation of fire risks, safety critical applications of software-based technology, as well as human and organisational performance.

This final report summarises the goals and results of the programme. The programme has published 57 scientific articles, 233 mainly international conference papers, and 274 other reports. Six doctoral theses, two licentiate and 18 master's theses were completed. The total volume of the programme during the four years has been about 130 person-years and Euro 14.4 million.

Preface

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 technology programmes on nuclear fusion, advanced light water reactor concepts and plant life management, funded partly by the National Technology Agency (Tekes).

In 1998, 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 combined programme and for its organisation. The programme, *The Finnish Research Programme on Nuclear Power Plant Safety (1999–2002) FINNUS (Kansallinen ydinvoimalaitosten turvallisuustutkimusohjelma),* was to concentrate on the themes of ageing, accidents and risks. The concept of national research programmes will also continue after FINNUS: KTM has commissioned a long-term-plan for the next programme for the years 2003–2006, called preliminary SAFIR (Safety of nuclear power plants – Finnish national Research programme).

The Technical Research Centre of Finland (VTT) has co-ordinated the FINNUS programme and performed most of the research with a significant contribution from Lappeenranta University of Technology (LTKK). The main funding sources have been KTM, VTT, the Radiation and Nuclear Safety Authority (STUK), the Finnish power companies Teollisuuden Voima Oy (TVO) and Fortum Oyj. LTKK has offered a research laboratory for the use of the programme. These parties have also been represented in the steering group of the programme.

The execution of the programme has been based on the general plan and annual plans prepared for KTM in co-operation with the regulatory body, the power companies and the research bodies. This Final Report summarises the FINNUS programme. The report also includes summaries of the projects and a selection of more detailed results in the form of presentations of the FINNUS Final Seminar 14–15 November 2002 in Otaniemi, Espoo. Information on publications, international connections and academic degrees awarded have also been included.

This report was prepared by the programme leader and the project co-ordinator, in co-operation with the project leaders and members of the programme staff. Several persons from various institutions have actively contributed to the steering, strategy and reference group activities of the FINNUS programme. This support is greatly appreciated by the programme staff.

Contents

1. Introduction

A continuous high level of safety is a prerequisite for the use of nuclear energy. About 27 % of electricity generated in Finland is nuclear, and in 2002 the Government made a positive decision-in-principle, approved by Parliament, concerning the construction of a new nuclear power plant (NPP) unit. High operational reliability and careful upgrading have maintained competitive production costs also in the open electricity market. Confidence in nuclear safety calls for continuous investment in plant operation and supervision. These activities are effectively supported by well directed research in various fields of technology and human behaviour, in order to take into account modernisation and upgrading of plant processes, implementation of new techniques, changing production goals and renewal of safety requirements. Research also deepens the understanding of new technology needed in the construction and operating licensing phases of new units.

In the planning of the research period of FINNUS, 1999–2002, 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 [1]. Thus, the previously separated fields of structural and operational safety research were combined into the Finnish Research Programme on Nuclear Power Plant Safety FINNUS. The Ministry of Trade and Industry (KTM) launched the programme and has administrated it. In order to guarantee independent know-how and research resources for the public sector, the needs of the Radiation and Nuclear Safety Authority (STUK) were dominant in the planning of the programme. This aspect has also been reflected in the funding and steering structure of the programme. In order to combine the limited national resources, the Finnish power companies have also contributed to the national programme by offering resources for specific tasks and by participating in the steering and reference groups.

The general objectives of the FINNUS programme in the long term were to
- develop tools and practices for safety authorities and utilities
- provide a basis for safety-related decisions
- educate new nuclear energy experts
- promote technology and information transfer.

The research objectives of the FINNUS programme were classified under three themes, **ageing**, **accidents** and **risks** aiming at integrity, safety and reliability. The effects of ageing on nuclear power plants (NPP) have been studied intensively in order to evaluate the safe remaining lifetime of the components and the efficiency of possible corrective measures. The ageing field covered material sciences of the metallic structures in a nuclear power plant, structural integrity studies and in-service inspection and monitoring methods including reinforced concrete structures as a new area. The accident theme covered operational aspects of nuclear power plant safety. Research has been conducted in the fields of basic reactor physics and dynamics, fuel studies, thermal hydraulics and special questions of severe accidents. 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 to the risk or reliability evaluation of a certain process or technology. In the FINNUS programme, the latter group included studies on fire risks, safety critical applications of software-based technology, as well as human and organisational performance.

Until the publication of this report, the FINNUS programme has produced a total of 564 reports in various categories given in Appendix A. A major publishing channel has been conference presentations but also 57 refereed scientific articles have been produced. Many detailed technical results were documented as working reports with limited distribution. Concurrently with this report, an Executive Summary report of the programme has been published [2].

The research methods have formed a broad spectrum:

- development of theoretical models, from technical to psychological
- use of highly sophisticated mathematical methods
- development of computer codes
- calculations and simulations with international and in-house computer codes
- studies on chemical and physical phenomena
- planning and construction of novel experimental facilities
- carrying out and analysing experiments and tests
- collection of statistical data, as well as
- use of observation and interviews when applicable.

International co-operation has been vital in all the fields of the programme – which has traditionally been intensive in the nuclear energy research area. The most important contact organisations have been the OECD Nuclear Energy Agency (NEA), the International Atomic Energy Agency (IAEA), the Commission of European Communities (EU) and the Nordic Nuclear Safety Research (NKS). NKS and the OECD Halden project financed some tasks in the programme. As a main rule, the projects of the EU Framework Programmes were not financially included in FINNUS. However, there has been lively collaboration between FINNUS and EU projects, not least of all thanks to the research scientists taking part in both.

In addition to publishing activities in various international forums, the research staff have contributed to working groups and networks, as well as defined and solved international benchmarks and participated in round robin exercises. The main contacts to international organisations and research units are summed up in Appendix B. The programme has been presented to national and international technical communities and nuclear safety organisations. In 2000, a mid-term seminar of the whole programme was arranged and an interim report was published [3].

The research programme contributed to education of new experts in the field of nuclear safety in co-operation with the universities. Six doctoral theses, two licentiate and 18 master's theses were completed. Their subjects are given in Appendix C. In 2002, the programme employed a total of 25 young researchers and research trainees participating in the research projects as summer trainees or with a purpose of graduating. On-going studies for licentiate and doctoral degrees in the projects include both young researchers and senior experts.

FINNUS (1999–2002) briefly:	
Research fields: NPP ageing, accidents and risks **Research Partners**: VTT Processes VTT Industrial Systems VTT Building Technology Lappeenranta University of Technology (LTKK) **Co-ordination:** VTT Processes	**Number of research projects: 11** **Total volume (1999–2002)**: Euro 14.4 million 130 person-years **Funding in 2002** (million Euro): KTM: 1.40 VTT: 1.00 STUK: 0.65 Utilities: 0.23 Others: 0.26
http://www.vtt.fi/pro/tutkimus/finnus/ or seek at **http://www.vtt.fi**	

In FINNUS there have been 11 research projects. Chapter 2 gives brief information on these projects and their main results, describing their interdisciplinary co-operation and providing some statistical information. In Chapter 3 a summary of the entire programme is given. In Chapters 4–14 the projects are presented in more detail. Each chapter includes a project summary report of the main goals and achievements as well as a selection of more detailed research results in the form of presentations of the FINNUS Final seminar Nov. 14th – Nov. 15th 2002 in Otaniemi, Espoo. KTM has commissioned a long-term-plan for the next national nuclear safety programme for the years 2003–2006, called preliminary SAFIR (Safety of nuclear power plants – Finnish national research programme) [4].

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2. FINNUS programme

2.1 Organisation

The research has mainly been performed by VTT with a significant contribution from Lappeenranta University of Technology (LTKK). VTT has also co-ordinated the execution of the programme. The general plans were prepared for the four-year period 1999–2002, and re-evaluated in the strategy seminar in 1999. The execution has been based on annual plans prepared by the project leaders and programme staff under the guidance of the project reference groups and accepted by the steering group [1–4]. The results have been summarised as annual reports [5–7].

In the organisation of steering and support functions of the programme, strong coupling was ensured between the funding bodies, research organisations and the end-users of the results, both at the nuclear safety regulator and at the power companies. The main duty of the steering group was to supervise the overall performance of the programme. The steering group set up reference groups for the research projects and nominated strategic planning groups on the main themes.

The project reference groups contributed to the planning of the projects, e.g. by suggesting new research topics. They advised on the projects and evaluated the results against the plans. An important task of the reference groups has been to communicate between the research groups and the users of the results.

The strategic groups evaluated the priorities and discussed the challenges of the whole programme against national needs, as well as contributed to planning the main contents of the next programme.

The steering group and the project reference groups typically met 3–4 times a year, while the strategic planning groups met twice during the programme. The organisation scheme of the FINNUS programme is shown in Figure 1. The list of persons involved in the steering and reference groups, as well as programme staff and their main duties are presented in Appendix D.

The steering and reference groups evaluated the FINNUS programme in connection with the mid-term seminar in 2000. With the focus also on the next programme, the projects and their results, their educational value, applications and targets were more thoroughly evaluated again in the beginning of 2002 by the reference groups. A final evaluation will be carried out in 2003.

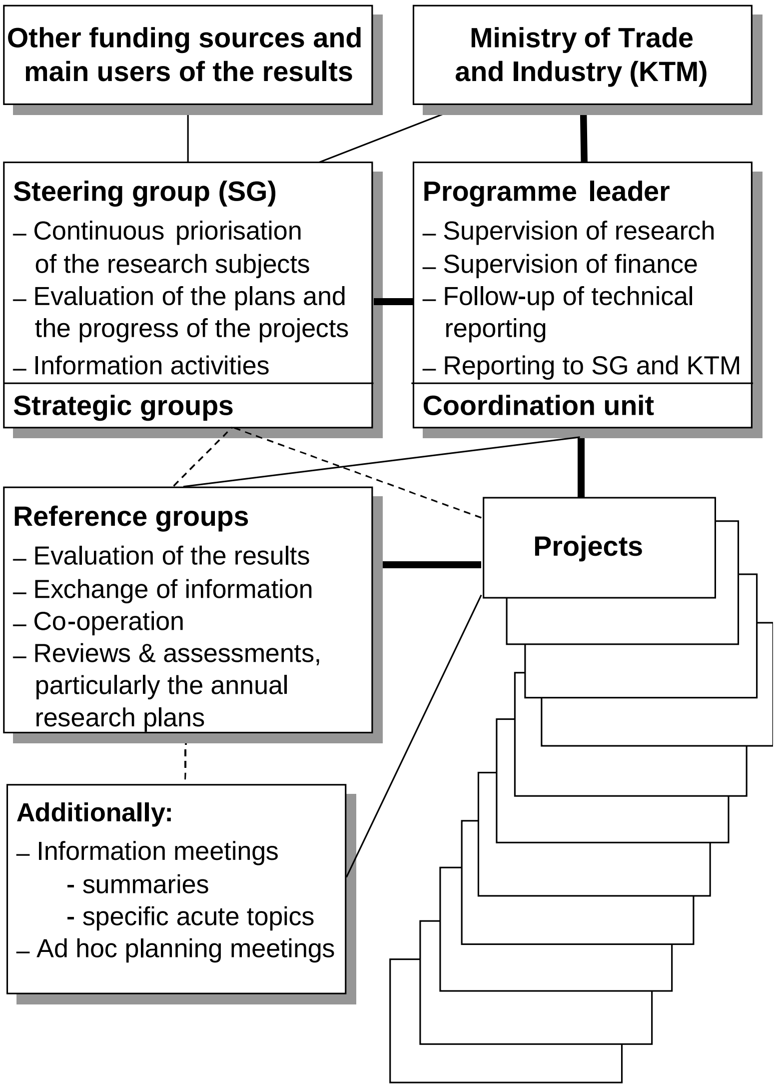

Figure 1. Organisation of the FINNUS programme.

In FINNUS there have been total of 11 research projects in the fields of ageing, accidents and risks, as indicated in Table 1. There were natural connections between the projects, and one objective of the programme was to strengthen these links. All the projects were planned for the four-year period, but detailed objectives varied during the execution of the programme. VTT has co-ordinated the FINNUS programme and also performed most of the research with a significant contribution from LTKK, which has also offered the thermal hydraulics research laboratory for the use of the programme.

Table 1. Names and research organisations of the FINNUS projects.

Main theme Project name (Project Acronym) / *Project name in Finnish*	Research organisations
Ageing	
Ageing phenomena (AGE) / *Vanhenemisilmiöt*	VTT Industrial Systems VTT Processes
Structural integrity (STIN) / *Lujuuden varmistaminen*	VTT Industrial Systems VTT Processes
In-service inspections and monitoring (INSMO) / *Tarkastukset ja kunnonvalvonta*	VTT Industrial Systems VTT Building Technology
Accidents	
Transient behaviour of high burnup fuel (KOTO) / *Korkeapalamaisen polttoaineen transientti- ja onnettomuuskäyttäytyminen*	VTT Processes
Reactor physics and dynamics (READY) / *Reaktorifysiikka ja -dynamiikka*	VTT Processes
Thermal-hydraulic experiments and code validation (TOKE) / *Termohydrauliset kokeet ja ohjelmistojen kelpoistus*	Lappeenranta University of Technology VTT Processes
Modelling and simulant experiments of severe accident phenomena (MOSES) / *Vakavien reaktorionnettomuuksien mallinnus ja simulanttikokeet*	VTT Processes VTT Industrial Systems
Risks	
Fire safety research (FISRE) / *Paloturvallisuus*	VTT Building Technology
Programmable automation system safety integrity assessment (PASSI) / *Ohjelmoitavan automaation turvallisuuden arviointi*	VTT Industrial Systems
Methods for risk analysis (METRI) / *Riskianalyysin menetelmät*	VTT Industrial Systems
Working practices and safety culture in NPP operations (WOPS) / *Toimintatavat ja turvallisuuskulttuuri ydinvoimalaitoksessa*	VTT Industrial Systems
Administration and information of the research programme (HALTI) / *Tutkimusohjelman hallinto ja tiedotus*	VTT Processes

2.2 Description of the projects and their main results

The research objectives of the FINNUS programme were classified under three themes, **ageing**, **accidents** and **risks** aiming at integrity, safety and reliability. The relationships of the projects to these themes are shown in Figure 2.

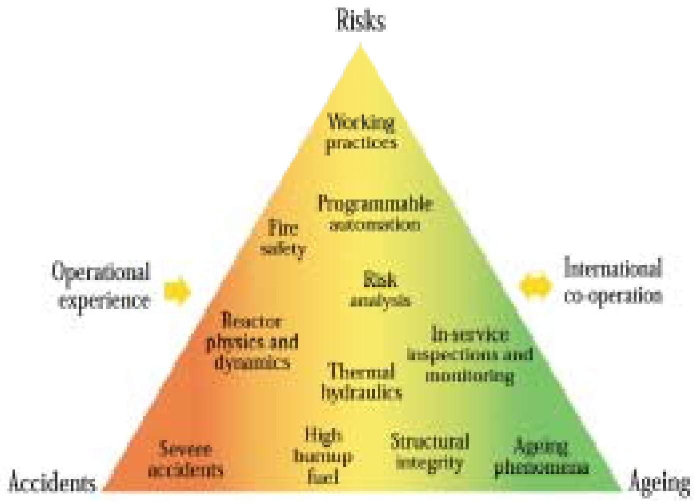

Figure 2. Locations of the projects in the "triangular research field" of FINNUS.

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.

The research in the **Ageing phenomena project AGE** was mainly oriented on ageing to ensure or to extend the operating lifetime of the components of nuclear power plants. New degradation mechanisms such as stress corrosion cracking of some weld metals and irradiated stainless steels have recently been discovered in aged materials. These latest ageing mechanisms are not included in the existing life management plans. Such degradation mechanisms observed in old facilities should also be avoided when planning new reactor units. 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 the 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.

In the AGE project, a bellows-loaded fatigue-testing capability has been created. Corrosion fatigue tests can be carried out to define the materials' lifetime and the nucleation of cracks. Rising displacement crack growth rate measurements can be conducted under LWR conditions and various interrelationships for environmentally assisted cracking can be understood. A Mixed Conduction Model (MCM) for the inner oxide layer on construction materials in primary circuits has been developed in order to understand how the water chemistry influences corrosion and activity incorporation. An experimental set-up has been constructed that enables electrochemical studies of construction materials even in low-conductivity high-temperature water such as BWR coolant, with the option to detect dissolving species as well. A model has been created to estimate the effects of irradiation/annealing cycles on the mechanical properties and on the re-embrittlement. Water chemistry, chemical composition of materials, corrosion potentials and oxide film as well as mechanical properties and loading parameters form an interacting circle determining environmentally assisted cracking phenomena in NPPs.

The main goal of the **Structural integrity project STIN** was to create verified experimental and computational methods, and to verify the existing methods for assessing the remaining lifetime of components and structures and their ability to withstand the possible accident situations. During the four-year project period, the main focus was on developing material characterisation methods, methods for defining loading conditions and developing fracture mechanism analysis methods.

The results obtained in the development of methods for small specimen toughness characterisation of irradiated materials led to major improvements in the reliability and material usage of fracture toughness testing. At the same time, the transferability of basic test results to structural integrity assessment was significantly improved. The computational tools and methods for assessing structures under thermal and impact loading conditions were developed and verified. Local approach fracture mechanics computation methods were found viable and worth pursuing in the predictions of material toughness as well as ductile crack propagation and brittle fracture initiation. A calibration methodology for linking the Master Curve method to a local approach with Weibull statistics was identified. The methodology is dependent on determining the normalisation fracture toughness and provides a nearly identical fracture toughness temperature dependency to that of the Master Curve.

The **In-service inspections and monitoring project INSMO** has concentrated on the techniques and systems that are applied to examine the structural integrity of critical components by non-destructive evaluation (NDE) methods. Additionally, the monitoring of material properties has been considered to some extent. The conventional non-destructive methods are usually applied to metallic components and materials. The reinforced concrete structures have been an interesting additional area, which has brought together people with different background competence. New problems from the standpoint of the conventional NDE have been brought out. Based on the measurement tests performed, it seems possible to find solutions to many of these challenges.

Methods to produce suitable artificial reflectors for the qualification samples of ultrasonic inspection have been examined and these reflectors have been measured and analysed. Two computer codes for simulation of ultrasonic inspection have been installed and applied in the project, as well as to some extent in practice. A new method based on spectral analysis of the ultrasonic signals has been designed and programmed in order to offer future possibilities to measure material characteristics.

Accidents

The accident field covers fuel research, reactor physics and dynamics, experimental and computational thermal hydraulics and severe accidents, which have been studied in the **KOTO, READY, TOKE** and **MOSES** projects.

Reactor analyses become more and more demanding for several reasons: higher discharge burnups are pursued, best-estimate type approach is increasingly favoured, and acceptance criteria assessments are becoming more sophisticated. The purpose of the **Transient behaviour of high burnup fuel project KOTO** has been to keep in pace with this development as far as fuel performance modelling is concerned.

The project supports assessing the licensing criteria and evaluating the consequences of efforts for improved fuel utilisation. Statistical methods for fuel analyses have been elaborated for extensive applications. Two parallel steady-state and accident code pairs are now well established and in operation at VTT: the USNRC codes FRAPCON-3 and FRAPTRAN, and independently, ENIGMA and SCANAIR. A new development with the READY project, featuring advanced thermal hydraulics coupled with a fuel accident performance model, the FRAPTRAN-GENFLO code, is opening an unparalleled capability for realistic transient simulation.

Additional high burnup data for materials and for rod integral behaviour are available from recently launched international programmes. The Finnish organisations are very well placed in the most essential of these, with the arrangement of participation and application mainly managed by VTT. High burnup effects must not be overlooked as regards postulated accident behaviour. STUK has lately granted the utilities a new assembly burnup limit of 45 MWd/kgU for the fuel types now in use. The methods and expertise due to the KOTO project were among those that supported both STUK and the utilities in assessing this upgrade.

In the **Reactor physics and dynamics project READY** and in its predecessors a computer code system and competence has been created and maintained for carrying out all reactor physics calculations needed in Finland. Additionally, a comprehensive and independent computer code system and expertise for reactor safety analyses have been developed, thus providing tools from basic nuclear data to three-dimensional transient and accident analyses.

As a result of upgrading, adapting, developing and validating several reactor physics and dynamics codes of VTT's code system, more accurate calculations and predictions of physical phenomena can now be performed. Three-dimensional reactor physics codes have been introduced and validated for various applications, especially for out-of-core flux, criticality safety, and dose rate calculations. A multi-temperature MCNP cross section and scattering law library have been created. Most recently, the Monte Carlo technique has been applied in burnup calculations. The development of a new, advanced nodal model has resulted in a highly promising new BWR simulator code, ensuring independent computational capabilities for steady-state and safety analyses. Validation of VTT's three-dimensional BWR dynamics code has been completed, thus enabling analyses with three-dimensional core models to be carried out for BWR, PWR and VVER type reactors. Until now, the transient analyses for BWRs have been done with one-dimensional models. Fuel models of the dynamics codes have been improved, and USNRC's fuel behaviour code has been coupled with VTT's advanced hydraulics model in co-operation with the KOTO project. The coupled code has been delivered to the USA in agreement with USNRC. The development and application of a sophisticated hydraulics solver in reactor dynamics has resulted in a BWR circuit model that has been tested in steady-state calculations.

The **Thermal-hydraulic experiments and code validation project TOKE** addressed both the experimental and computational aspects of nuclear safety studies. Integral VVER-related experiments dealing with a steam generator collector header rupture incident and with non-condensable gas behaviour in the primary circuit were carried out in the Parallel Channel Test Loop (PACTEL). Local loading effects due to water flow and thermal stratification in a T-joint of a hot horizontal pipe and a cold vertical tube were investigated in a purpose-built test loop in co-operation with the STIN project. The behaviour of non-condensable gas during the first seconds of a conceivable large break loss-of-coolant accident (LBLOCA) blowdown to a BWR condensation pool was also studied in a subproject related to separate effect tests. For this purpose, a test rig with a scaled down water pool, blowdown pipes, an emergency core cooling system (ECCS) strainer and a pump were designed and constructed in LTKK. Thermal hydraulic and computational fluid dynamics (CFD) calculations with the codes APROS and Fluent, respectively, supported the planning and analysis of both the integral and separate effect tests.

The **Modelling and simulant experiments of severe accident phenomena project MOSES** has systematically investigated phases of severe reactor accidents relevant to Finnish NPPs. The areas of research have been pressure vessel failure mode, core debris coolability, fission product behaviour, including chemistry, containment thermal hydraulic loading, especially hydrogen detonations and phenomena relevant for long-term severe accident management. The outcome of the project has been novel analysis tools, procedures and tools to perform multi-disciplinary analyses and new experimental results reducing noticeably uncertainties that existed in the field of severe accidents.

A new FEM-model was developed and validated against experiments for 2D- and 3D-analyses of pressure vessel lower head creep behaviour. The model is suitable for assessing failure of lower head with or without penetrations. The coolability of particulate core debris in the containment water pool has been assessed with simulant experiments. The particle bed characteristics were chosen to be representative to the Olkiluoto plant. In collaboration with the STIN project, a capability of analysing consequences of supersonic, energetic hydrogen combustion has been developed comprising 3-dimensional simulation of thermal hydraulic loads, and analysis of the structural response of concrete walls and steel pipelines to the loads. Long-term severe accident management issues have been studied by evaluating the radiation levels in the containment atmosphere and water pools, and the effects of radiation on the water pool chemistry. The results can be used to assess human accessibility into the containment and the survivability of various accident management devices in the high radiation field.

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** was organised into three subprojects: effect of smoke and heat on electronics, modelling of fire scenarios for probabilistic safety assessment (PSA), and studies on active fire protection equipment. The strategic goal of the FISRE project was to improve the quantitative capability of fire-PSA in NPPs. Using technology transfer, experiments and modelling of phenomena, a new in-depth understanding of fire problems has been obtained. Furthermore, new measuring and smoke detection techniques have been proposed. The methods developed find wide application also outside the nuclear field, as shown by examples of recently constructed buildings. A Monte Carlo calculation platform was made for estimating probability distribution for the time of damage of the second train in a cable tunnel of a NPP, given the ignition of the first train. Realistic data were collected from utilities for both the cable tunnels and an electronics room, and both cases were calculated. The tool has been released outside VTT. Literature studies, experiments and modelling lead to a quantitative understanding of the acute effects of smoke, humidity or both, on the loss of insulation resistance of control electronics in NPPs. A quantitative calculation formula for the effectiveness of protective coating was proposed.

In general, the **Programmable automation system safety integrity assessment project PASSI** aims to provide support for the authorities and utilities in the licensing process of programmable automation systems. In the PASSI project, new reliability assessment methods and practices for software-based systems have been developed and tested. The key assessment method developed and tested in the project is based on Bayesian statistics and in particular its technical solution called Bayesian networks. Bayesian networks enable the implementation of reliability evidence from disparate sources providing at the same time a consistent way of reasoning the assessor’s beliefs on the relationship of the different kinds of evidence. The method was tested in an experimental application study on a qualitative reliability estimation of a software-based motor protection relay. The results gained on the use of the method were encouraging and further testing of the methodology will be practised in the benchmark exercise next year. A study on the failure mode and effects analysis of software-based systems was carried out and the results of the study will be utilised in the future example cases. Information related to the ageing of modern instrumentation and control (I&C) equipment was gathered and a preliminary test plan for the simulation of certain operational ageing was prepared.

The **Methods for risk analysis project METRI** focused both on PSA methodology and its applications, especially in multi-criteria decision-making situations where PSA results are combined with other, deterministic criteria to select an optimal decision alternative. The project has promoted the use of risk-informed approaches by developing methods for PSA qualification and expert panels to support risk-informed decision-making. Application areas are the optimisation of in-service inspections, risk-informed safety classification of systems and components, and the reliability of advanced passive systems. The main topics concerning PSA methodology were model uncertainties and human reliability analyses.

Along with the increasing use of PSA in safety-related decision-making, the development of decision analytical methods and tools to facilitate the risk-informed regulation and safety management has become important. The integration of different expertise in the decision process calls for formal approaches in order to achieve a balanced result. It is also important to verify the quality of PSA in relation to the decision in question, so that an appropriate weight can be given to the PSA results in risk-informed decision-making. The METRI project has addressed topics related to the reliability analysis of passive systems, the analysis of human errors of commission, and methods for uncertainty analysis.

The **Working practices and safety culture in nuclear power plant operations project WOPS** has concentrated on human and organisational factors. The project focused on two main topics, the development of the competencies of control-room operators and the organisational culture in high-reliability organisations. Within these topics, several studies were concluded in which practically relevant problems of the Finnish NPPs were tackled. Thus, the studied issues were transformation of operational expertise in a generation change situation, constraints on control-room operators in fire situations, organisational culture in the Nuclear Reactor Regulation department of STUK, and organisational culture in the maintenance department of the Loviisa NPP. As a background of the latter study, an analysis of human errors in maintenance was conducted. New empirical results were obtained in these studies. The work also contributed to a long-term methodological aim to develop a new framework for the analysis of human conduct in dynamic, complex and uncertain situations.

2.3 Co-operation between the projects

The achievements of the projects have been summarised in the previous chapter. There were natural connections between the projects and one objective of the programme was to strengthen these links. In Figure 3, the projects and their connections during the FINNUS programme are shown.

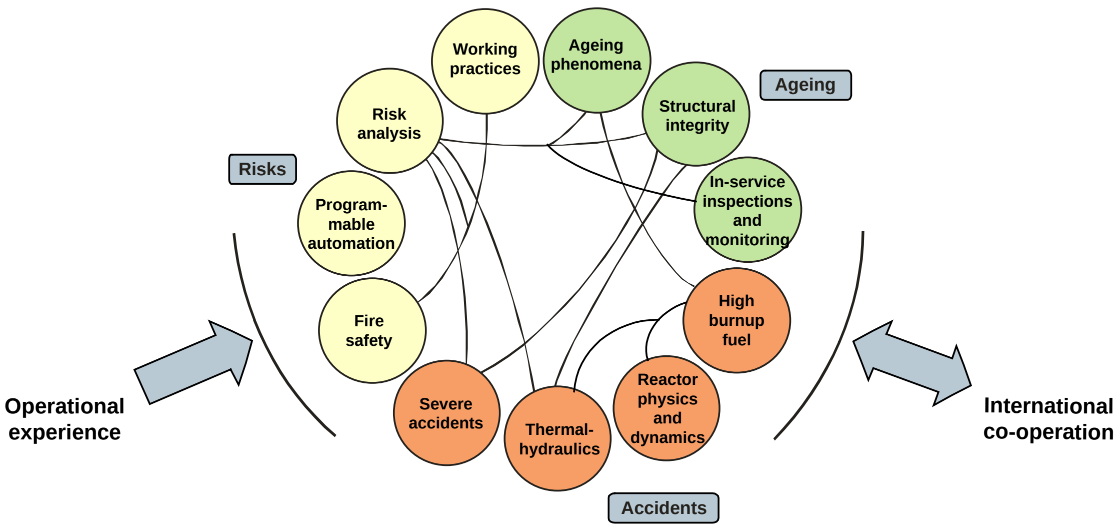

Figure 3. The 11 research projects under the three themes of the FINNUS programme. The projects associated with a certain theme are strongly inter-coupled. Couplings materialised during the programme, especially between the themes are indicated. Also the connections with the end-users have been strengthened in many projects during the programme.

The links in Figure 3 describe the following tasks:

Thermal hydraulics / Structural integrity: experiments and calculations on local loading effects due to thermal stratification in a T-joint, and design calculations for condensation pool experiments.

Severe accidents / Structural integrity: hydrogen detonation pressure loads on the containment structures during severe accidents.

Risk analysis / Severe accidents: phenomenological uncertainties.

Risk analysis / Thermal hydraulics: passive safety system reliability.

Risk analysis / Ageing phenomena, Structural integrity, In-service inspections and monitoring: risk informed in-service inspection, and investigation of failure modes of degraded pressure equipment, i.e. structural reliability.

Working practices / Fire safety, Risk analysis: method for analysis of management of fire situations in NPPs.

Ageing phenomena / High burnup fuel: corrosion and material phenomena of high burnup fuel.

High burnup fuel / Reactor physics and dynamics: coupling of USNRC's fuel performance code with VTT's thermal hydraulics code that is utilised also in simulators and severe accident calculations.

In addition, other types of co-operation have emerged within the programme framework. Long-term severe accident management issues have been studied both by chemists and physicists. The experience of the material research unit of VTT has been utilised also in the construction and performance of the tests on the severe accident experimental facility. Nuclear power plant buildings have been surveyed to identify possible problem areas where non-destructive inspection or measurement methods used in the primary circuit inspections could be applied. One important function of this task has been the familiarisation of the specialists of building technology with nuclear constructions and the co-operation of NDE specialists from different fields.

A small country has limited national resources for research. However, the well-organised nuclear community has enabled novel couplings between different research fields extending the know-how of the scientists. The external experts participating in the work of the reference groups of the projects have also benefited from the interdisciplinary approaches and extensions of the projects. People with different background competence have worked in co-operation.

The programme has been actively presented to national technical communities and nuclear safety organisations. In 2001 and 2002 various seminars were arranged by the projects individually or in co-operation with each other, see Table 2.

Table 2. Seminars arranged in FINNUS projects during 2001–2002.

Project	Date	Title	Part.
AGE	30/31.1.01	Seminar on water chemistry and oxide film chemistry	~ 20
	28.5.01	Irradiation damage & oxidation in copper and copper alloys	13
	30.10.01	Risk-informed ageing management of structures *	28
	7.2.02	Seminar on water chemistry and oxide film chemistry	~ 20
STIN	12.6.01	Structural integrity	~ 15
	30.10.01	Risk-informed ageing management of structures *	28
INSMO	11.6.01	Inspection & monitoring	~ 10
	30.10.01	Risk-informed ageing management of structures *	28
KOTO READY	10.10.01	Nuclear fuel day	35
TOKE	5.2.02	Experiments on thermal stratification and condensation pool behaviour in LTKK and related calculations at VTT	27
MOSES	24.9.01	Modelling of severe accident phenomena and simulant experiments	22
	12.9.02	OECD / OLHF experiments and analyses	16
FISRE	10.9.02	Latest fire safety engineering - from NPPs to public assembly halls	60
PASSI	12.3.02	Seminar on programmable automation system safety integrity assessment	45
METRI	30.10.01	Risk-informed ageing management of structures *	28
WOPS	25.10.01	Maintenance of know-how: generation change in nuclear power plants	22
HALTI	20.8.02	Strategic seminar on planning of the next programme	60

* Co-operation seminar of METRI, AGE, STIN and INSMO projects

2.4 Statistics

VTT has performed most of the research with a significant contribution from LTKK. The main funding sources have been KTM, VTT, STUK, TVO and Fortum. The main part (about 4/5) of the costs comprises salaries. The annual volume and funding of the projects are shown in Table 3. The total volume of the programme resources has remained fairly even during the four-year period.

Table 3. Annual volume and funding of the projects.

Project (Acronym)	**Volume** (person-years)				**Funding** (thousand Euro)			
	99	00	01	02 plan	99	00	01	02 plan
Ageing phenomena (AGE)	5.3	4.4	4.5	3.1	661	567	628	528
Structural integrity (STIN)	2.7	3.7	2.8	3.0	342	436	397	415
In-service inspections and monitoring (INSMO)	1.2	1.3	0.9	0.8	167	160	148	123
Transient behaviour of high burnup fuel (KOTO)	2.4	1.9	1.6	1.7	215	188	170	182
Reactor physics and dynamics (READY)	6.7	5.4	5.2	4.7	547	495	512	487
Thermal-hydraulic experiments and code validation (TOKE)	4.4	3.3	4.2	4.0	514	445	411	466
Modelling and simulant experiments of severe accident phenomena (MOSES)	3.3	2.5	2.3	1.7	445	412	431	366
Fire safety research (FISRE)	1.0	1.1	0.9	1.3	102	141	117	115
Programmable automation system safety integrity assessment (PASSI)	1.7	1.7	2.0	1.7	137	175	231	222
Methods for risk analysis (METRI)	2.3	2.4	2.4	2.1	247	224	268	251
Working practices and safety culture in NPP operations (WOPS)	2.2	2.5	2.5	2.0	188	293	227	266
Administration and information of the research programme (HALTI)	0.9	0.7	0.7	0.9	118	96	97	120
Total	**34.1**	**30.9**	**30.0**	**est. 30.0**	**3683**	**3578**	**3639**	**3541**

Until the publication of this report, the FINNUS programme has produced a total of 564 reports in various categories listed in Appendix A. In Table 4, the number of publications is arranged project by project as also in the appendix. A major publishing channel has been conference presentations but also refereed scientific articles have been produced. Many detailed technical results were documented as working reports with limited distribution.

Table 4. Publications of the projects.

Project	Scientific publications	Conference papers	Research institute reports	Others	Total
AGE	21	47	4	16	**88**
STIN	11	34	2	16	**63**
INSMO	–	8	1	12	**21**
KOTO	–	11	–	–	**11**
READY	2	31	6	54	**93**
TOKE	2	13	–	18	**33**
MOSES	3	11	11	34	**59**
FISRE	2	26	1	15	**44**
PASSI	–	4	2	6	**12**
METRI	10	24	12	23	**69**
WOPS	6	20	12	17	**54**
HALTI	1	4	4	8	**17**
Total	**58**	**233**	**55**	**219**	**564**

The research programme contributed to education of new experts in the field of nuclear safety in co-operation with the universities. Six doctoral theses, two licentiate and 18 master's theses were completed. Their distribution among the projects is shown in Table 5 and their subjects are listed in Appendix C. In 2002, the programme employed a total of 25 young researchers and research trainees participating in the research projects as summer trainees, or with a purpose of graduating. On-going studies for licentiate and doctoral degrees in the projects include both young researchers and senior experts.

Table 5. Academic degrees awarded in the projects.

Project	**Doctor** (DTech, PhD)	**Licentiate** (LicTech, LicPhil)	**Master** (MScTech, MSc, MA)	Total
AGE	2	–	1	**3**
STIN	–	–	1	**1**
INSMO	–	–	–	**–**
KOTO	–	–	1	**1**
READY	–	1	4	**5**
TOKE	1	1	3	**5**
MOSES	1	–	1	**2**
FISRE	–	–	5	**5**
PASSI	–	–	1	**1**
METRI	2	–	–	**2**
WOPS	–	–	1	**1**
Total	**6**	**2**	**18**	**26**

2.5 Nuclear energy research in Finland

The total volume of nuclear energy related R&D efforts in Finland in 1999 were about Euro 27 million, see Figure 4. No dramatic changes have occurred since then. The power companies directly fund more than half of the total volume, and the public sector about one third. Half of the total volume is spent on 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 allocated for fully or partly public research programmes.

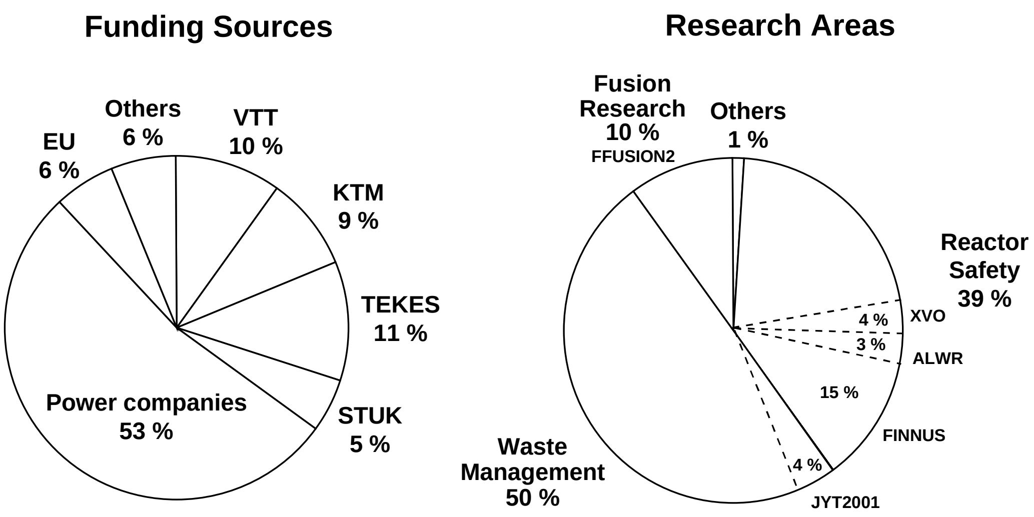

Figure 4. Resources of nuclear energy research in Finland in 1999. The public funding comes from the Ministry of Trade and Industry (KTM), the Technical Research Centre of Finland (VTT), the National Technology Agency (Tekes) and from the Radiation and Nuclear Safety Authority (STUK).

The largest of the public research programmes is FINNUS, which has concentrated on nuclear reactor safety related issues of the existing power plants. The Advanced Light Water Reactor programme ALWR has dealt with possible future solutions of nuclear power generation. The Plant Life Management Programme XVO was directed towards plant-specific ageing problems with particular support from the power companies. JYT2001 was the public nuclear waste management programme. These programmes have mainly been conducted at the various research units of the Technical Research Centre of Finland (VTT). Universities have also contributed to these programmes. Similar programmes have been continued further; in 2002 the National research programme KYT for waste management and the project RKK for ageing were started. KTM has also commissioned a long-term-plan for the next national nuclear safety programme for the years 2003–2006, called preliminary SAFIR (Safety of nuclear power plants – Finnish national Research programme).

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3. Summary

The main motive for public nuclear safety research is to guarantee independent resources for the Finnish regulatory body in the safety evaluation of nuclear power production. This aim includes the development of tools and practices, the 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 13 years, various fields of nuclear safety research have been organised as national research programmes, which have been launched and administrated by the Ministry of Trade and Industry. The Finnish Research Programme on Nuclear Power Plant Safety FINNUS (1999–2002) was conducted mainly at the Technical Research Centre of Finland (VTT) and Lappeenranta University of Technology (LTKK) and co-ordinated by VTT Processes.

In FINNUS, the fields of structural integrity and operational safety and reliability have been combined into three main themes of ageing, accidents and risks. A total of 11 research projects have been conducted under these themes. The effects of **ageing** on nuclear power plants have been studied intensively in order to evaluate the safe remaining lifetime of the components and the efficiency of the corrective measures. The programme has mainly concentrated on studies in ageing effects on material properties and degradation mechanisms of metallic structures, structural integrity and in-service inspection as well as monitoring methods, including reinforced concrete structures as a new area. The **accident** theme has concerned the operational aspects of nuclear power plant safety. The issues of nuclear fuel behaviour, reactor physics and dynamics modelling, thermal-hydraulics and severe accidents were addressed under the theme by conducting both computational and experimental studies. In the **risk** field, attention has been paid to advanced risk analysis methods and their applicability, and to the evaluation of fire risks, safety critical applications of software-based technology, as well as human and organisational performance.

The annual volume has been about Euro 3.6 million and 30 person-years. Until the publication of this report, the FINNUS programme has produced a total of 564 reports in various categories. A major publishing channel has been conference presentations but also 57 refereed scientific articles have been produced. Many detailed technical results were documented as working reports with limited distribution.

International co-operation has been vital in all the fields of the programme. In addition to publishing activities in various international forums, the research staff have contributed to working groups and networks, as well as defined and solved international benchmarks and participated in round robin exercises. The programme was presented to national and international technical communities and nuclear safety organisations.

The research programme contributed to the education of new experts in the field of nuclear safety in co-operation with the universities. Six doctoral theses, two licentiate and 18 master's theses were completed. In 2002 the programme employed a total of 25 young researchers and research trainees participating in the research projects as summer trainees or with a purpose of graduating. On-going studies for licentiate and doctoral degrees in the projects include both young researchers and senior experts.

In the steering and selection of research goals of the FINNUS programme, the Radiation and Nuclear Safety Authority (STUK) has had a dominant role. 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 contributed to funding of selected research topics of the FINNUS programme.

In FINNUS there were natural connections between the 11 research projects, and one objective of the programme was to strengthen these links. The nuclear community of a small country has been an advantage when creating novel couplings between different research fields. The interdisciplinary approaches of the projects have extended the know-how of both the research scientists and the external experts participating in the work of the reference groups.

4. Ageing phenomena (AGE)

4.1 AGE summary report

Pertti Aaltonen
VTT Industrial Systems

Introduction

Main research on ageing is oriented to ensure or to extend the operating lifetime of the components of nuclear power plants. The plant life extension was technically established and described in mid 1980's. However, new degradation mechanisms have currently been discovered in aged materials and thus, these latest ageing mechanisms are not included in the existing life management plans. On the other hand, degradation mechanisms observed in old facilities should be avoided when planning new reactor units.

The current ageing mechanisms of components exposed to reactor environments are studied 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) and irradiation induced swelling observed in austenitic stainless steels even in highly reducing environments, such as those in PWR's.
- Intergranular or interdendritic cracking of nickel base weld metals.
- Dynamic strain ageing of nitrogen alloyed ferritic steels as well as low carbon austenitic stainless steels (nuclear grade materials).

Additionally, corrective actions taken in the nuclear power plants to prevent or mitigate the degradation caused by the ageing mechanisms have generated new open questions, such as:

- 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 lead containing 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 preventive 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.

Main objectives

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 [1]. 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 or replacement time.

The targets and goals for the materials research related to ageing in the FINNUS programme have been defined along the lines of international programs [2] as follows:

- 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 power plants, to establish understanding of ageing mechanisms involved and to transmit information from the international programmes related to ageing. The research topics dealing with concrete, electric components or cables were not carried out within the AGE project but they were covered to some extent by other FINNUS projects.

Environmentally assisted cracking of nuclear materials (ENVI)

The Selective Dissolution-Vacancy-Creep (SDVC) model, initially presented for Alloy 600, has been further studied by tests conducted with copper base metals. According to the obtained and reported results, dissolution through the oxide film is a precondition for the corrosion deformation interactions through the vacancy generation in the material around the crack tip. The actual fracture advance event can be described by vacancy/dislocation interactions and recovery of the deformed crack tip material. Recovered material at the crack tip can be further deformed which enable further dissolution [2].

Autoclave testing has to provide enough data for statistical evaluations in the case of EAC phenomena, similar to fracture mechanics testing in air. Development of loading equipment capable of testing several specimens simultaneously while keeping environmental parameters constant, is the way to provide enough data for the application of statistical analysis methods, which is needed both for crack initiation and growth rate evaluations.

The capability to measure crack initiation in highly stressed materials and to re-assess the original fatigue life calculations using time consuming crack initiation measurements in autoclaves has been improved. Additionally, on-line monitoring of water chemistry parameters conducted at many plants, as part of ageing related monitoring programs, provides information of real environments to be used in laboratory fatigue initiation measurements. An enormous task of producing fatigue crack initiation data in relevant reactor environments has been started by conducting strain controlled tests in autoclave and reference tests in air for nuclear grade AISI 316 stainless steel [4].

The general need to provide more data points for statistical analysis has been acted by development of bellows loading rigs, which are able to load several specimens simultaneously. Crack growth measurements using three point bending specimens and low rising displacement rate testing in a bellows loading rig is a routine and strain controlled fatigue testing in reactor environments have passed all demonstration phases. Verification tests in air at room temperature, as well as at high temperature and high pressure have been successfully performed [5]. Bellows loading fatigue testing rig development has been continued by construction of module autoclaves, providing for each loading rig an own autoclave. The advantage comes through the flexibility to maintain or replace specimens while other tests keep running.

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

Understanding changes in the properties of oxide films that are formed on construction materials during plant operation and possibly due to irradiation is crucial with regard to ageing phenomena. 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 is aimed 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 focused 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 has been shown that the influence of zinc ions on Co-60 incorporation is due to the competition between zinc and cobalt for the adsorption site [6]. 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 focused on the transport of radioactive species in the outer, porous part of the oxide film. The first step was 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 focused 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 oxide layer 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 [7, 8]. The

MCM has also been applied to explain the behaviour of oxide films on Ni-Cr alloys in nearly-neutral conditions up to 200°C [9].

The need of experimental data to be 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 has been 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 has among the other things characterised factors affecting crack initiation through the interface between irradiated material and the coolant, i.e., oxide film.

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 have been conducted. Research on the theoretical bases of material behaviour in annealing and the effects of re-irradiation in the micro-mechanistic and macroscopic level has also been conducted.

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 [10]. 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 correlation 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. Reanalyses of existing data, supplementary tests with irradiated IAEA CRP materials, irradiation shift estimations and clarification of the inconsistent lower shelf values have been initiated in order to further verify the small specimen reconstitution technique [11].

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 [12]. 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 Field Emission Gun Scanning Electron Microscope (FEGSTEM) to find out the role of phosphorus in embrittlement of the irradiated model alloys has been conducted 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 the activity build-up. However, evaluation of the effects of these new water chemistry approaches on the fuel cladding materials requires more information. The tendency towards more severe operating conditions induced by the need for longer service times and higher fuel burn-ups has also called for more information through characterising the relationship between the composition and microstructure of Zr alloys, their oxide growth kinetics and susceptibility to local corrosion modes. Despite the fact that there is a generally agreed qualitative picture of the oxidation process of zirconium alloys in high-temperature water, the exact mechanism of conduction through the oxide has remained largely unidentified.

The main result of the present work is that the in-situ electrochemical impedance spectroscopy (EIS) measurements of Zircaloy-2 in high-temperature water corresponding to typical waterside Boiling Water Reactor conditions at 300°C were performed successfully using a Controlled Distance Electrochemical (CDE) arrangement. The EIS measurements provided important information on the electrical and electrochemical properties, the thickness and in-homogeneity of the oxide films formed on Zircaloy-2 fuel cladding materials.

The contribution of the oxide film to the impedance spectra was distinguished from the kinetic contribution of the corrosion reaction by carrying out impedance measurements in an inert gaseous atmosphere, in addition to the in-situ measurements in simulated BWR water. A fit of these two types of EIS data to the Mixed Conduction Model made it possible to obtain a correlation between the impedance parameters, film thickness and oxide growth rate on Zircaloy-2. This allowed to calculate the oxide growth rate, which is assumed to be proportional to the flux of oxygen vacancies in the oxide film [13]. The results indicate that the properties of the oxide layer do not change as a function of time after the barrier layer has been formed. The present work demonstrates that electrochemical impedance spectroscopy is a fast and useful tool for in-situ corrosion rate measurement of zirconium alloys.

International co-operation

There are several organisations working on 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.

Within the AGE project the research on ageing phenomena of materials in nuclear power plants is conducted in close co-operation with the international programs. The main goal of the FINNUS AGE project is to understand the mechanisms of ageing and the consequences of degradation for the components. Thermal degradation, corrosion, fatigue, radiation embrittlement of construction materials and all their combinations have been of major concern.

Applications

The main types of applications are:

- Fatigue data generation jointly within international co-operation.
- Surveillance testing of irradiated reactor material applying testing with sub-size, possibly reconstituted, specimens.
- On-line monitoring of water chemistry transients in plants and estimation of consequences.
- Risk informed NDE inspections.

Conclusions

Water chemistry, chemical composition of materials, corrosion potentials and oxide films formed under specific conditions as well as mechanical properties and loading parameters form an interacting circle determining environmentally assisted cracking phenomena in NPP's.

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4.2 Ageing and oxidation of reactor materials

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Abstract

The target of this project has been to study and model the changing properties of ageing pressure boundary materials used in nuclear power plants. Models to estimate and methods to measure the behaviour of these materials under various operational conditions in nuclear power plants have been developed and applied. In nuclear reactors stainless steels are mainly used as a material facing the primary circuit environment. Thus, the main focus has been to study the role of the primary circuit environment and surface oxide films containing Fe, Ni and Cr on changes occurring in the mechanical properties and degradation susceptibility of ageing pressure boundary materials.

Introduction

Ageing of pressure boundary materials in nuclear power plants takes place due to loads, temperature, irradiation and environment. The changing properties of materials in various components may facilitate sensitivity to such degradation mechanisms which were not accounted for in the design phase of the plant.

Metallic alloys, used as construction materials in nuclear reactor components, form a protective oxide film on the alloy surface as a result of reactions with the environment. Without formation of such a film, the alloys would suffer general corrosion. Besides the mechanical properties of materials, also the properties of oxide films, e.g., their structure and composition, are changing, influencing the susceptibility to localised forms of corrosion. In addition, in nuclear power reactors oxide films have a great influence on other material properties, such as heat transfer, the incorporation of active species on the primary circuit surfaces, sizing and friction of surfaces etc. Minimising the risks of degradation of ageing pressure boundary materials calls for understanding of the ageing mechanisms of metals and oxides. To understand the possible consequences of these changes experimental research of material properties and environmental parameters i.e., water chemistry has been conducted. Further, to understand the various combined effects due to ageing on mechanical properties and oxide films of construction materials, models and testing methods of these interactive ageing phenomena are being developed. In the future these models and methods can be used partly also for predicting the structural integrity of plant components, corrosion phenomena as well as the incorporation of activity on surfaces.

The work carried out in each subproject of the Ageing phenomena (AGE) project carried out as a part of the Finnish Research Programme on Nuclear Power Plant Safety (1999–2002) FINNUS programme consisted of three common goals:

- mechanistic verifications and modelling of specific ageing mechanisms
- development of improved experimental research techniques and
- international co-operation combined with the education of experts.

This paper contains summary of the progress made during the AGE project in FINNUS programme on verification and modelling of ageing mechanisms and development of improved experimental techniques within ENVI and OXI subprojects.

Environmentally assisted cracking of nuclear materials (ENVI)

Continued development of the quantitative model "Selective Dissolution – Vacancy Creep" (SDVC) for environmentally assisted cracking

The idea of Selective Dissolution-Vacancy-Creep (SDVC) model, initially presented for Alloy 600, included selective dissolution of iron, production of vacancies due to selective dissolution as well as deformation localisation to a shear band. In the SDVC model the cracking is believed to be in connection with the generation of vacancies by selective dissolution of cations through the existing passive film. According to the model, dissolution through the oxide film has a major effect on the vacancy generation in the material around the crack tip. The actual fracture event is described by vacancy/dislocation interactions. The SDVC model has been applied in the present work to pure copper in order to generalise the model. Copper has been chosen because it provides all relevant cracking phenomena at ambient temperatures and the oxide structure is more simple compared to stainless steels.

Vacancy generation through dissolution of copper

Vacancies are injected into brass due to preferential dissolution of zinc. The generation of vacancies due to selective dissolution or oxidation in brass takes place when the metal is passivated, i.e., covered with an oxide layer of cuprous oxide, Cu_2O. The oxide reactions at passive potentials have been described in the point defect model (PDM) by Macdonald and in the mixed-conduction (MCM) model developed in the OXI subproject (see below). Based on these models, changes in the environment, e.g. increase of the concentration of surface active anions or alloying with elements entering into the oxide film with higher oxidation state or external anodic polarisation may increase the cation vacancy flux through the oxide barrier layer and, thus, increase the vacancy generation in the underlying metal. In the cuprous oxide barrier layer the charge carriers are electron holes. The amount of holes in cuprous oxide is increased by anodic polarisation. Anodic polarisation produces an increase in the amount of electron holes, i.e., Cu^{++} ions in the oxide film formed on pure copper. Anodic polarisation increases the flux of cation vacancies through the barrier layer until the potential is high enough to form cupric oxide CuO associated with reduction in the cation flux. Electroneutrality calls for an increase in the amount of cation vacancies as a result of increased electron hole concentration. This, in turn, facilitates the transport of cations through the film and thus accelerates dissolution providing finally more metal vacancies at the film metal interface according to the following equations:

$$\Delta(h) \cong \Delta (V_M^-) \quad (1)$$

$$M_M \rightarrow M^{\delta+}{}_{(aq)} + V_M^- + \delta \cdot e^- \quad (2)$$

$$m + V_M^- \rightarrow M_M + V_m + e^- \quad (3)$$

where V_M^- = cation vacancy in the oxide, h = electron hole in the oxide, M_M = metal cation in the cation site in the oxide, V_m = vacancy in the metal and m = metal atom in metal.

Vacancy Dislocation Interactions And The Crack Advance Event

In front of the crack tip the material is highly strained, which means that the dislocation density is very high corresponding to applied loading. The vacancies react with the dislocations facilitating recovery of the crack tip material reducing dislocation density. Recovered crack tip material can be deformed further by the applied external loading.

Pressure vessel steel studies

Applicability of small specimen techniques and elastic-plastic approach for determining stress corrosion cracking (SCC) susceptibility and crack growth rate were the main interests in pressure vessel steel studies. Ferritic pressure vessel steel was used in this study. Fracture resistance curves (constant displacement rate tests) with three constant displacement rates $1...2 \cdot 10^{-7}$ mm/s, $4 \cdot 10^{-7}$ mm/s, and $6...8 \cdot 10^{-7}$ mm/s were determined using small SEN(B) specimens under simulated BWR conditions with 10 ppb and without sulphide SO_4^{-2} addition. The dimension of the SEN(B) specimens were 10 x 10 x 55 mm^3 (thickness x width x length) and all specimens were precracked so that the ratio of initial crack length over width, a_0/W was 0.5. Tests were performed displacement controlled using pneumatic bellows loading system which enabled six simultaneous tests in an autoclave.

The results confirmed that presence of sulphur at the crack tip initiating either from the MnS inclusions of the steel or from the BWR coolant favours stress corrosion cracking. Susceptibility for SCC was revealed and SCC crack growth rates obtained from small specimen constant displacement rate tests were in accordance to the test results obtained with 1' or 2' C(T) "large" specimens presented in the literature.

Constant displacement rate provides increasing displacement to the test specimen and eventually creates elastic plastic loading conditions in small test specimens. Although the ligament of the small test specimen will be plastically deformed, the process zone at the crack tip where the actual crack growth takes place is equal to the large specimens' process zone at the same values of stress intensity factor K or J-integral. It seems that the "excess" plasticity in ligaments of small specimens do not have strong contribution on SCC crack growth rate. On the other hand, increasing displacement during the test provides strain and strain rate in the crack tip. This causes oxide film rupture and enables SCC to occur. This means that the constant displacement rate tests (slow fracture resistance curve determinations) reveal SCC susceptibility more easily than constant load or constant displacement tests.

Fatigue

The first autoclave tests using the VTT Bellows fatigue units was performed in PWR water, 320 °C for AISI 316 NG [1]. Ramp loading was used with a frequency of 0.1 Hz for 0.25 % strain amplitude and 0.01 Hz for 0.4 % strain amplitude.

This first data for 316NG steel indicates a notable effect of PWR environment (320 °C). However, the data lies above the ASME III design curve in good agreement with the data published by Argonne National Laboratory (ANL), as can be seen from Figure 5, where a comparison is made to ANL test data for 304, 316 NG and cast CF 8M stainless steels in different environments [2].

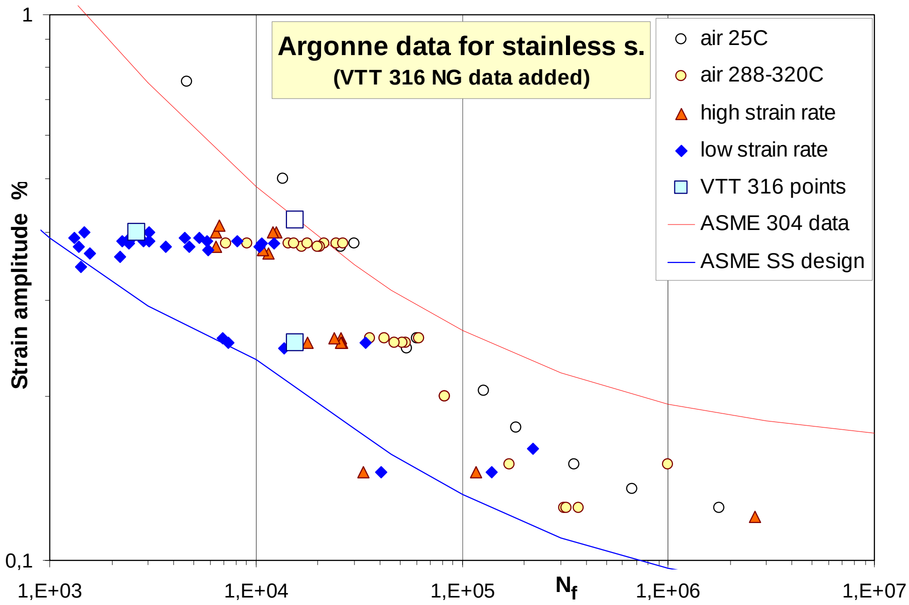

Figure 5. VTT corrosion fatigue data for AISI 316 in PWR conditions compared to the ANL test data [2] for 304, 316 NG and cast CF 8M stainless steels in different environments .

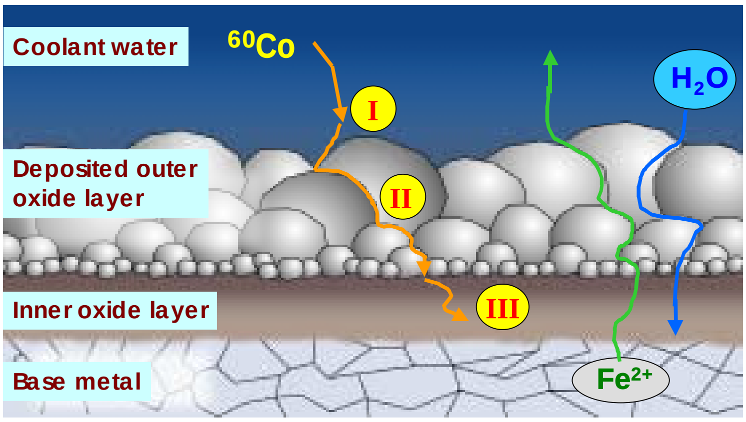

Figure 6. A scheme of the incorporation of active species in the oxide film.

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

Description and role of oxide films on construction materials

The aim of the OXI subproject has been to increase understanding of the electrochemical and chemical behaviour of construction materials in different operational conditions. The main focus has been on activity incorporation and corrosion reactions in high-temperature aqueous environments.

Incorporation of radioactive species from the coolant on the primary circuit surfaces may lead to a need to decontaminate plant components. To influence the incorporation rate of radioactive species, it is essential to understand the mechanism and to identify the rate-limiting steps of their incorporation and also of the corrosion of the component materials. This in turn requires modelling of the processes taking place in the oxide films formed on component surfaces.

An oxide film forming on stainless steel in typical power plant conditions is shown schematically in Figure 6. The outer, deposited part of the oxide film is porous, while the inner part is more compact. The incorporation of radioactive species into the oxide layers may be controlled by surface phenomena (step I in Figure 6), by the transport of species in the outer layer (step II) or by the transport of species in the inner layer (step III). The incorporation rate is different for different incorporating species.

The experimental techniques developed during 1997–1998 [3] and models to describe the adsorption on oxide surfaces and processes in different parts of the oxide film at room temperature have made it possible to start the research in conditions corresponding to NPP coolants as summarised below. The goal of the work has been to model the processes in different parts of the oxide film influencing the activity incorporation and corrosion in high-temperature conditions. As a starting point, the film has been assumed to remain in a steady state, which means that its thickness and composition do not change significantly with time.

Adsorption by surface complexation

The first step in the interaction of an active species, like ^{60}Co, with the oxide film on stainless steels is adsorption (step I in Figure 6). When ^{60}Co in the coolant arrives at the oxide film surface it occupies a surface site and may form a complex molecule. Competitive adsorption of less harmful species may reduce the rate of activity incorporation, which pinpoints the important role of the coolant composition as well. Injection of zinc has for instance been observed to lead to a decreased rate of activity incorporation. The surface complexation approach [4] has been applied in the present work to elucidate the effects of solution conditions on adsorption. The approach represents adsorption in terms of interaction of the adsorbate with the OH-groups of the oxide resulting in the formation of specific surface complexes. The OH-groups are able to release protons (H^+) thereby creating attractive adsorption sites on the surface.

Both the modelling approach developed in the present work and the establishment of experimental facilities for high-temperature adsorption studies have made it possible to start estimating the relative adsorption of such additions as Zn, Ni, Fe and Mn in primary circuit conditions. Model calculations [4] have already demonstrated that the retarding effect of zinc on the adsorption of cobalt on iron oxide can be theoretically predicted. At this stage, the model can predict competitive adsorption phenomena semiquantitatively. Finding and assessing relevant thermodynamic data for quantitative modelling of adsorption phenomena at elevated temperatures has been one of the main tasks during the project. The model has been experimentally verified by means of high-temperature titrations of hematite using different cations.

Transport in the porous oxide layer

Adsorption of an active species on the surface is followed by its transport into the outer porous layer of the oxide film (step II). The properties of the outer oxide layer influence the transport rate of radioactive species towards the inner, compact layer. The thickness of the outer layer in typical simulated BWR conditions is a few hundred atomic layers [5] while its thickness in typical simulated PWR conditions is smaller [6]. In real plant conditions with a long history including several transients the oxide film thickness may be considerably higher [7].

A mathematical model has been developed to describe the transport of active species in the high-temperature aqueous water inside the pores of the outer oxide layer [8, 9]. The model includes general high-temperature aqueous phase and oxide compositions, adsorption by way of surface complexation in the pores and co-precipitation as well as multi-species reactions for ions, aqueous complexes, precipitates and dissolved gases.

The results have made it possible to estimate, for instance, the potential drop within the pores of the porous oxide layer. The potential drop contributes to the establishment of a possible driving force for the transport of radioactive species. The results have also indicated that the morphology, i.e. the size and geometry of the crystals in the porous part of the film, affects the transport of radioactive species. The morphology may in turn be affected by water chemistry, for instance by the Fe/Ni ratio in the coolant.

Transport in the compact oxide layer

The compact 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. In special cases such as within a stress corrosion crack, the protective properties of the compact oxide film may be strongly connected with the susceptibility to environmentally assisted cracking (EAC) and other forms of local corrosion.

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 defect structure of the film may also greatly influence which radioactive species are able to enter the film, how fast they may be transported through the film and how strongly they may be occupy a position in the film. The mixed-conduction model (MCM, see below) and the experimental facilities developed in the present work have made it possible to assess the defect structure of films formed in different conditions. Thus they have greatly improved the possibility to treat the role of the compact layer in activity incorporation and corrosion. In order to correlate more quantitatively the properties of the compact layer with water chemistry, a combined model of the adsorption and the phenomena in the porous and compact parts of the film is required.

The MCM model [10] explains 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. In the present work, the MCM has been extended to high-temperature films and applied to experimental data on films formed on Fe-Cr, Fe-Cr-Mo and Ni-Cr model alloys, pure Fe, pure Ni and pure Cr as reference materials and on commercial alloys [11–20]. The main outcome parameters from the MCM model include the conductivity profiles and mobility (and thus also the diffusion coefficients) of ions, electrons and vacancies in the film. The diffusion coefficient of cation vacancies dictates the transport rate of vacancies through the compact oxide film and thus also the transport rate of active species in the compact oxide layer. Verification of the model with this respect is on a semi-quantitative level at this stage.

Because the transport of cation vacancies towards the metal/film interface is likely to influence the initiation of stress corrosion cracking and crack growth, the ideas of MCM are worth combining with material properties and cracking models. In the near future it is foreseen that the effect of such properties of the underlying base metal as the strain and stress will be included in the MCM model.

Experimental method development

In-core Pd reference electrode

An in-core Pd reference electrode has been developed for the monitoring of corrosion and redox potentials in plant conditions. The sensor has been installed into the pressure vessel of the Halden test reactor in July 2002. The behaviour of the sensor has been compared with that of a commercially available GERS in-core magnetite electrode and of a Pt electrode developed by OECD Halden Reactor Project. During the two-week test period under radiation at high temperature the in-core Pd electrode provided reliable and stable potentials whereas the magnetite electrode failed mechanically after two days of operation. Also the Pt electrode showed some unexpected features. Radiation and high oxygen content of the water did not affect the potential of the in-core Pd electrode demonstrating that this type of sensor can be used as a reference electrode in in-core locations. The next test period will start in November 2002 and should last for two months giving information about the long-term stability of the in-core Pd electrode.

Sub size specimen testing methods

Experimental clarification of specimen size effects for EAC crack growth rate measurements under elastic plastic loading conditions in LWR environments has been performed. It was shown that the SCC susceptibility could be revealed with small specimen (10 x 10 x 55 mm^3) constant displacement rate tests. The SCC crack growth rate determined in the tests was evaluated using superposition analysis method. The crack growth rate results were in line with large specimen constant load and constant displacement tests. Effect of loading rate on EAC susceptibility as well as on crack growth rate under simulated BWR conditions has been analysed and will be reported, too.

Tensile and fracture resistance tests for cold worked austenitic stainless steel AISI 304 has been performed with different size tensile and fracture mechanical specimen. The results showed that small specimen (1 x 2 x 8 mm^3) tensile tests give similar yield stress and tensile strength values to those measured with standard size tensile specimens. Total elongation values measured with small specimens were clearly higher than the elongation values of standard specimens. In fracture resistance curve determinations small 8 x 10 x 55 SEN(B) specimens yielded similar J-R curves than standard 25 mm C(T) specimens with crack growths up to 30 % of initial ligament.

Development of bellows loaded many-specimen fatigue rig to be used under strain controlled fatigue tests in an autoclave

A pneumatic bellow loaded fatigue unit for performing 4 axial fatigue tests simultaneously in LWR autoclaves was designed and tested. The axial fatigue unit provided controlled reversed loading in a strain control mode with a maximum cyclic frequency of 0.5 Hz. Figure 7 provides a schematic figure of the bellows loading system used to provide fatigue loading. Using feedback from the pressure transducer near the bellows, the servo controller and servo valves control the load system air supply to meet the specified loading history: constant load, constant amplitude or spectrum. Also loading transients from reactor operation can be simulated. Depending on the test speed requirements 1 or 2 servo valves and metering valves can be used. For long term fatigue testing, a two-valve system will probably be used also to add redundancy and reliability.

First strain controlled corrosion fatigue limit data points for AISI 316NG stainless steel in PWR conditions has been measured and reported [1].

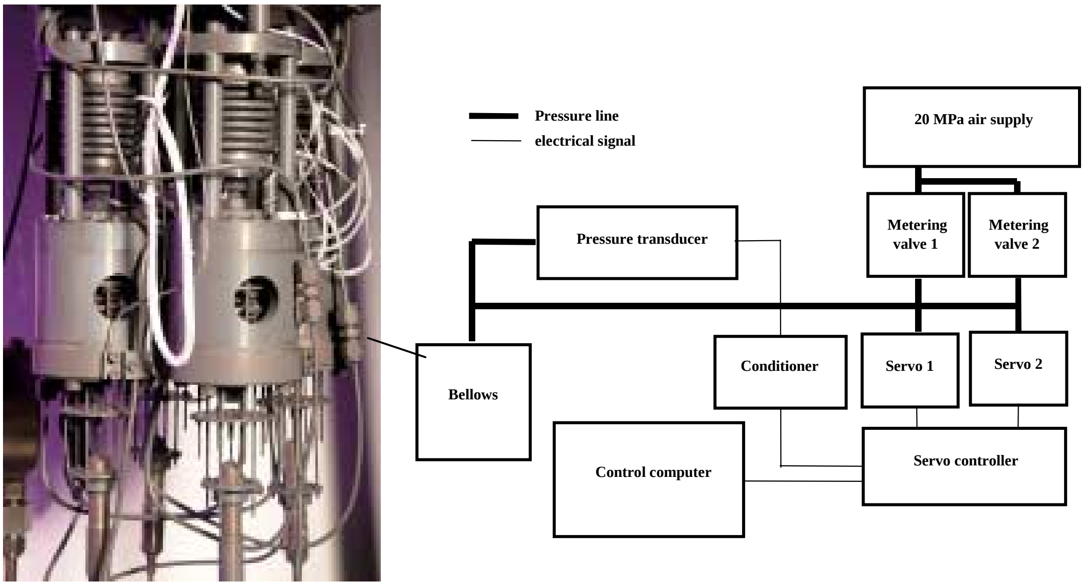

Figure 7. Bellow loaded 4-site unit for axial corrosion fatigue tests under LWR conditions.

Electrochemical techniques for high-temperature measurements

The need for experimental data to be used as input for the modelling of oxide films has led to 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. Therefore the controlled-distance electrochemistry (CDE) arrangement has been developed in this project [21]. This technique allows several different measurement options which can be summarised as follows:

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 [22] to investigate and to monitor the electronic properties of surface films.
- Contact electric impedance (CEI) measurements [23] to measure the solid contact impedance spectra of oxide films.

In order to be able to perform electrochemical tests under in-core conditions in a test reactor, a bellows-driven CDE arrangement has been developed and tested.

In addition, a photoelectrochemical set-up for high-temperature work has been developed and tested at temperatures up to 300 ^{O}C. The results have shown that oxides growing on iron-based materials show no photoactivity above 200 ^{O}C. Therefore, this technique does not provide the needed information on oxide properties relevant to operating NPPs, and its further development has not been pursued.

Conclusions

Environmentally assisted cracking (EAC) phenomena such as stress corrosion cracking (SCC) and corrosion fatigue (CF) of NPP components are affected by synergistic effects of environment, material and tensile stresses. Applied water chemistry and the chemical composition of construction materials determine the corrosion potential of the components in NPP coolants. The corrosion potential together with the composition of the coolant in turn influence the structure of the oxide films formed on component surfaces and also the extent of adsorption on the film surfaces. The properties of oxide films again influence the rate of activity incorporation and environmentally assisted cracking phenomena in a specific environment. Tensile stresses provide necessary strains and strain rates for EAC occurrance. These stresses derive from cyclic, quasi static or monotonically increasing loadings addressing to the components.

The Selective Dissolution-Vacancy-Creep (SDVC) model for stress corrosion cracking as well as more detailed models concerning the role, properties and behaviour of surface oxide films have been developed in the present project. The models can be applied to assess the influence of process parameters such as oxygen content, hydrogen content, Fe/Ni ratio, anionic impurity content etc. on oxide film behaviour and in the future to predict activity incorporation and corrosion suscebtibility.

Additionally, advanced SCC and CF test and analysis methods have been developed and demonstrated for investigations under LWR conditions. Rising displacement rate test method for EAC susceptibility and crack growth rate determination as well as VTT bellows fatigue units for strain controlled fatigue tests provide tools for reliable crack growth and fatigue life data generation.

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4.3 Re-embrittlement behaviour of steels

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VTT Industrial Systems

Abstract

The FINNUS programme has offered resources to participate in international joint projects in the field of material properties evolution during re-irradiation. Re-irradiation behaviour of the core weld is a generic open question of the old, annealed VVER-440 plants. The model alloy study is reported to some extent as an example of the performed studies.

Introduction

Many of the VVER-440 reactor pressure vessels have been annealed as a plant life extension measure due to their excessive irradiation embrittlement, including Loviisa 1. The annealing option was not considered during plant design and hence no surveillance programme for annealing and re-embrittlement was introduced in the beginning. Annealing complicates the estimation of the vessel mechanical properties, because limited number of material data on material behaviour in annealing-reirradiation cycles was available at the time of annealing and because the number of material parameters specifying the material condition increases greatly, i.e. material history will be a function of the irradiation, annealing and reirradiation parameters. In irradiation studies the relevant environmental parameters can seldom be repeated, the most problematic parameter being the irradiation time i.e. the effect of fluence rate on material behaviour [1].

The FINNUS programme has enabled the performance of generic studies in the field of material response to irradiation and re-irradiation. The studies have been realised mostly as participation in international co-operative programmes, the basic ones being the *Study of Model Alloys* in co-operation with JRC-IE Petten, Russian Research Centre-Kurchatov Institute, Fortum Nuclear Services Ltd and VTT [2] and the IAEA Co-ordinated Research Programme *"Round Robin Exercise on WWER-440 RPV Weld Metal Irradiation Embrittlement, Annealing and Re-embrittlement"* with 9 international participants [3].

Embrittlement can be measured with different toughness parameters, which are only partly commensurable. Hence the creation of data for the development of property to property correlations in different material conditions has also been an essential part of this subproject. The correlations are not discussed in the current paper. In re-irradiation studies the shortage of material has required the use of small specimens. One indispensable tool in practical data creation has been specimen reconstitution, which has also taken an important position in the subproject. This work will not be discussed either.

Modelling of embrittlement and re-embrittlement

Embrittlement modelling is based on the development of trend curves, i.e. the CH-V impact test based transition temperatures are described as a function of neutron fluence. CH-V test based transition shifts describe in the ASME practice the effect of irradiation on fracture toughness but fracture toughness can be measured also directly with irradiated specimens. Many regulators, countries and power companies have developed their own irradiation trend curves and these curves may have a formal status (law). The trend curves are a combination of a chemistry factor A_f and a neutron fluence dependent term.

The re-irradiation behaviour is conventionally described by purely experimental descriptions, which are shown in Figure 8. Because the original trend curves have been formally defined, the re-irradiation behaviour is based on them.

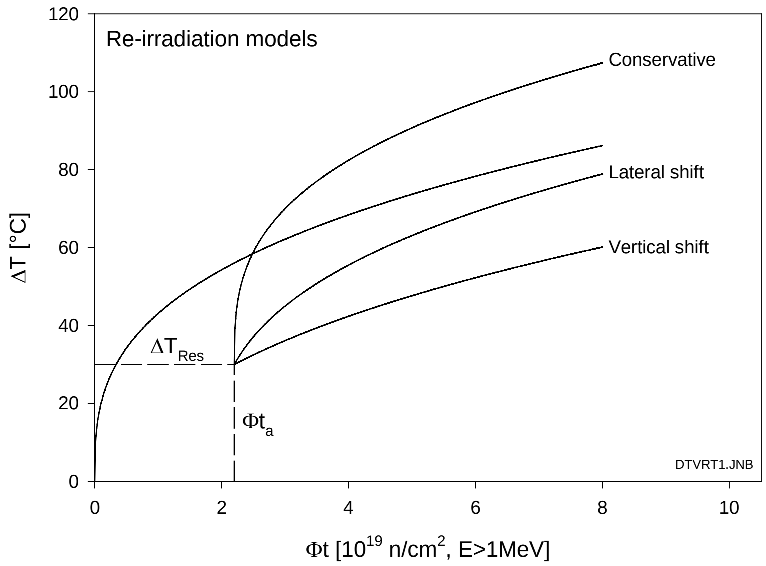

Figure 8. The vertical shift, lateral shift and conservative shift of re-irradiation models.

In the conservative shift model it is assumed that the embrittlement rate after annealing is the same as in the beginning with non-irradiated material. The lateral and vertical shift models are constructed by a clip and shift method from the original trend curves.

The elements in steels, which are active in embrittlement and re-embrittlement, are identified by the trend curve chemistry factors. Large part of the material property changes can be accounted for to copper, phosphorus and nickel contents. The identification of the physical formations is currently only on qualitative level. Copper and nickel containing formations have been identified by Field Ion Atom Probe Microscopy and by Small Angle Neutron Scattering. No clear formations responsible for the phosphorus effect have been identified even if phosphorus is the main embrittlement causing element in old VVER-440 steels.

Study of model alloys

Model alloys studies [2] were realised by materials let to be prepared by RRC-KI, by original irradiation performed in HFR in Petten and by re-irradiation in the Loviisa surveillance position. VTT tested the specimens in the irradiated, annealed and re-irradiated conditions. The Cu, P and Ni contents of the alloys were varied widely and altogether 33 alloys were prepared. Hence they allow a critical analyses of chemistry factors to be assessed. Sub-size CH-V specimens, which are cheep to test and which require small irradiation space, were used in the study. Figure 9 shows the fit of no-nickel data to the measured shifts.

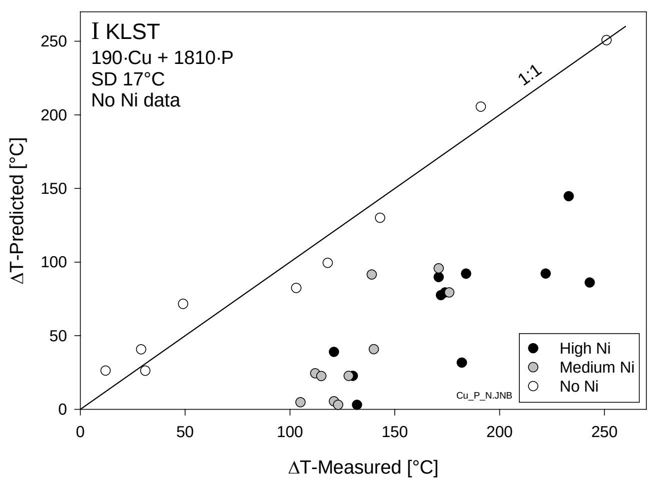

Figure 9. A linear copper and phosphorus fit to no-nickel data. Nickel data is also shown in the figure.

The chemistry factor derived from experimental data [Eq. 4] and the chemistry factor in the Russian norm [Eq. 5] are near each others.

$$A_{f,\ experimental,\ E>0.5MeV} = 800 * (1.24*P+0.13*Cu) \qquad (4)$$

$$A_{f,\ normi,\ E>0.5MeV} = 800 * (P+0.07*Cu) \qquad (5)$$

The best fit to whole data is given in Figure 10. The nickel exponent has a relatively low power (n=0.3), which means that the nickel effect switches on fast, when the nickel content increases, and it saturates soon. The fit to the re-irradiation shift data is shown in Figure 11. The derived functions, which describe the irradiation and reirradiation shifts, are given in [Eq. 6] and [Eq. 7]. The reirradiation neutron fluence was $1.0 \times 10^{19} n/cm^2$, E>1MeV as compared to the original irradiation fluence of $0.3 \times 10^{19} n/cm^2$, E>1MeV.

$$\Delta T = 167 * Cu + 1354 * P + 107 * Ni^{0.30}, \quad SD = 21\ ^{o}C, \quad \text{in irradiation} \tag{6}$$

$$\Delta T = 160 * Cu + 2852 * P + 99 * Ni^{0.33}, \quad SD = 23\ ^{o}C, \quad \text{in re-irradiation} \tag{7}$$

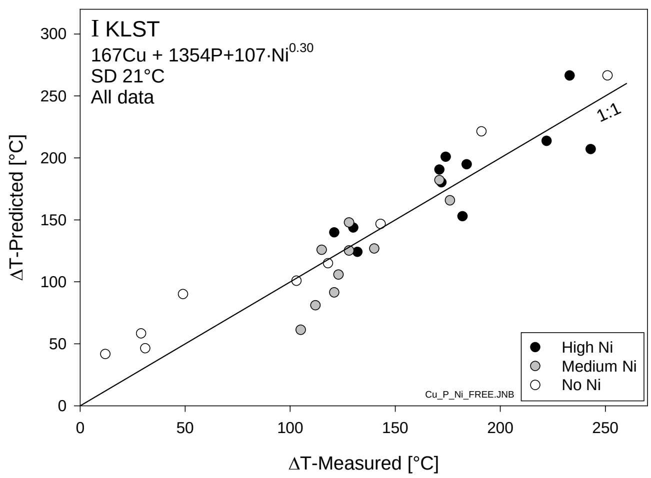

Figure 10. The best fit description of irradiation embrittlement shifts of model alloys. All four coefficients were free parameters in the fit.

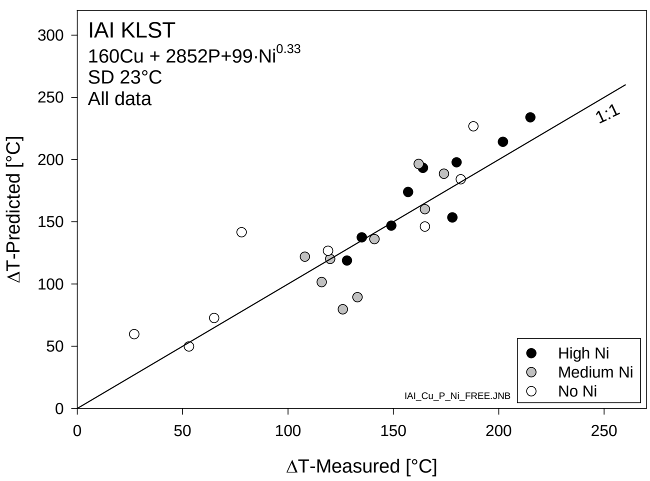

Figure 11. The best fit description of re-irradiation embrittlement of model alloys. All four coefficients were free parameters in the fit.

The fracture surfaces of the specimens were characterised by SEM lately and it was found that relatively large amount of intergranular fracture (IG) was noticed in the specimens in the irradiated and re-irradiated conditions but no IG-fracture in the annealed condition. IG-fracture is not typical for steels.

The measured re-irradiation behaviour compared to the original irradiation behaviour varied from the conservative shift description in the no-nickel, low copper and high phosphorus parameter corner to slower than the lateral shift behaviour.

Other studies

The focus in the IAEA co-ordinated research programme [3] was in the development of correlations between different toughness parameters as well as in the characterisation of irradiation, annealing and re-irradiation behaviour of the weld 502. Figure 12 shows the irradiation shifts measured with different types of specimens. It is clear from the figure that the use of sub-size specimens may lead to misleading estimations.

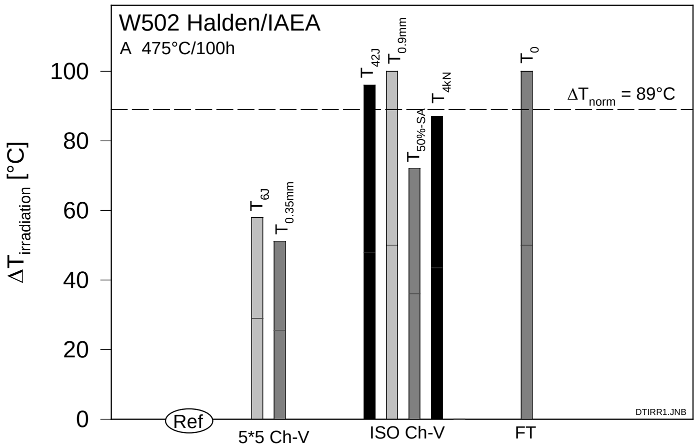

Figure 12. Irradiation embrittlement shifts measured with 5x5 CH-V, ISO CH-V and fracture toughness specimens.

The irradiation data may have rather large scatter especially, when data on different materials, specimen and test types and neutron fluence rates are analysed jointly. Figure 13 shows re-irradiation data, where on purpose data on different VVER-440 welds is jointly analysed. Curves in Figure 14 derived from this data indicate that the measured *average* re-irradiation behaviour is slower than the lateral shift prediction.

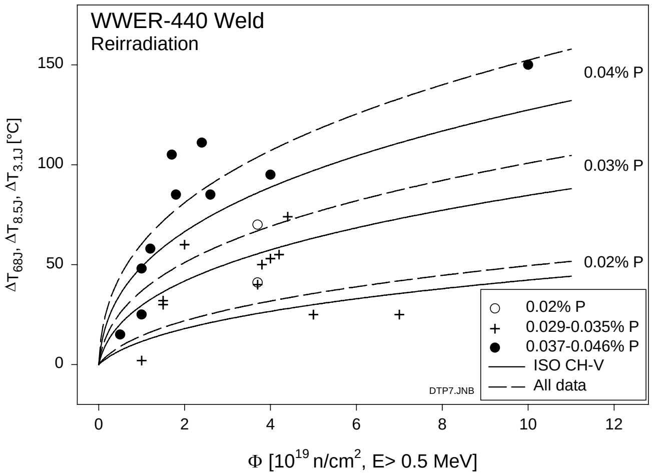

Figure 13. Model reirradiation functions derived from the whole data and the ISO CH-V data are shown as a function of phosphorus content and neutron fluence together with measured data. Specimen sizes and types, neutron fluences and fluence rates as well as the origin of material and their phosphorus contents vary.

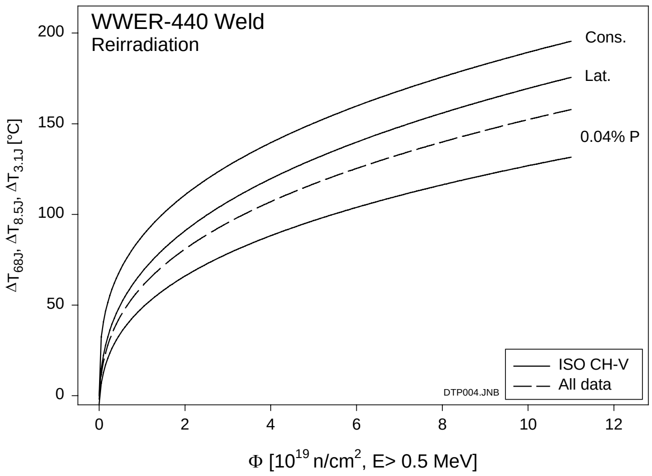

Figure 14. The average re-irradiation behaviour derived from data fitting (0.04%P) is compared with the upper limit lateral shift model (Lat.) and the upper limit conservative shift model (Cons.).

Conclusions

Pressure vessel annealing and the need to study material toughness evolution after annealing are a non-planned operation and its consequence. Currently the understanding of material re-embrittlement behaviour is not at the required quantitative level. The basic re-embrittlement micromechanism due to phosphorus, which is the main embrittlement causing impurity in old VVER-440 steels, has not been identified and consequently not modelled and verified. The modelling of mechanical properties as trend curves includes also much uncertainty assumably due to the variety of used materials, specimen types and differences in the irradiation and re-irradiation conditions. Proper understanding of re-embrittlement phenomena requires relatively large re-irradiation data base with varying material impurity contents and irradiations performed in well controlled conditions. Joint research is required between the VVER-440 owner countries and other interested parties in order to achieve this goal.

The role of the irradiation history in re-embrittlement requires also further clarification. This is especially important, because relatively large irradiated material stock has been created and this material has been used for re-irradation studies. Due to the large lead factor in VVER-440 surveillance the applied fluence and fluence rate parameters are relatively far away from the vessel exposure parameters.

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5. Structural integrity (STIN)

5.1 STIN summary report

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Abstract

The main goal of the Structural Integrity (STIN) project was creation of verified experimental and computational methods and also verifying the existing methods for assessing the remaining lifetime of components and structures and their ability to withstand the possible accident situations. During the four year project period, main focus was put on the development of material characterisation methods, methods for definition of loading conditions and development of fracture mechanism analysis methods.

Introduction

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. For the structural analysis and assessment, reliable input data is needed and different choices for the analysis methods are available depending e.g. on the most probable fracture mode, Figure 15.

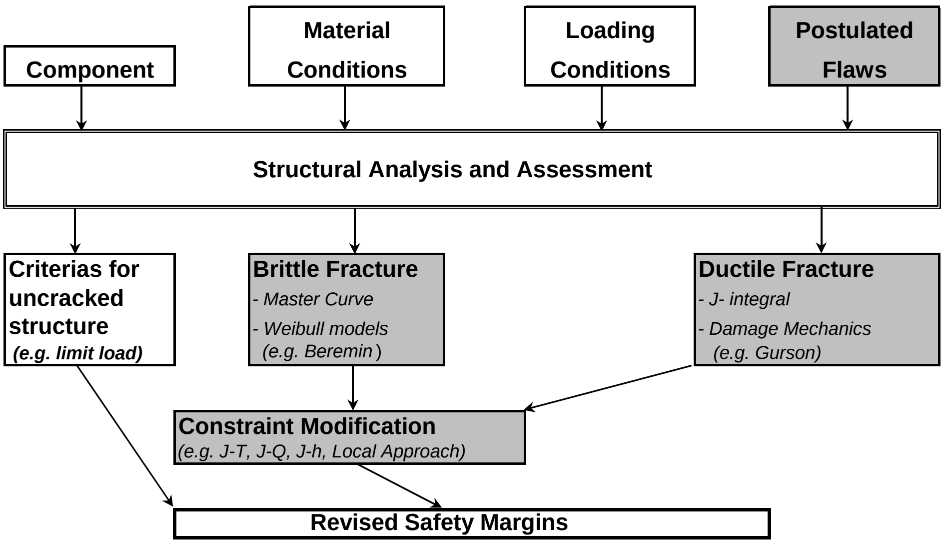

Figure 15. The top-down flowchart of the structural analysis and assessment.

The material properties should be reliably characterised using preferably small specimens. The existence of material inhomogenities in weld joints and in other discontinuities of geometry gives rise to specific demands on the modelling of the fracture mechanisms. Advanced 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. Advanced methods are also needed for modelling local loading conditions such as impacts and thermal stratification. These loads are calculated by special purpose codes or computational fluid dynamic codes. Loading transients are transferred to structural analyses using transfer tools created within this project. Structural analyses are carried out by using the finite element method.

Main objectives

The main objectives were i) creation of verified experimental and computational methods; ii) development of methods for reliable material characterisation using small specimens; iii) development of experimentally validated methods for the analysis of loading such as during thermal and pressure transients; and iiii) creation of understanding of the results by the improved micromechanical fracture models.

The project was divided into four subtasks: 1) Development of methods for small specimen toughness characterisation of irradiated materials; 2) Prediction of loading to structures and equipment under operational and accident conditions; 3) Fracture mechanisms of nuclear power plant structures and materials; 4) Validation of computational methods.

Development of methods for small specimen toughness characterisation of irradiated materials

At the beginning the main objective was to be able to derive reliable quantitative fracture toughness estimates from small material volumes and from minimum information of the material properties, and to derive quantitative crack arrest toughness estimates from brittle fracture initiation toughness properties. The development of correlations, e.g. for determining the fracture toughness from Charpy-V data was also an important aspect. Concerning the miniature specimens, a simple quality assurance methodology was later established to gain general acceptance for the use of miniature specimens, e.g. [1–2]. One of the key focuses was also on the development of the Master Curve approach to crack arrest applications, e.g. [3].

Results indicated that three point bend specimens may underestimate CT specimen fracture toughness contrary to fracture mechanics theory. A relationship between the

Master Curve fracture toughness and the T-stress, constraint correction, was developed, e.g. [4]. This enabled the application of the Master Curve Method also with low constraint geometries and explained also the difference between bend and CT specimens. Improvements to ASTM Master Curve standard were presented and the new version appeared in the book of standards 2002. Correlations, e.g. between J1mm (J-integral corresponding to 1mm crack growth) and Charpy-upper shelf level and between brittle fracture initiation and arrest toughness have also been developed, e.g. [5]. Also basic fracture mechanics testing methods and definitions were further developed [6–7]. Based on the work in STIN, concerning mainly miniature specimens, Kim Wallin was awarded the 2001 George R. Irwin Medal by ASTM, which is the highest technical recognition of fracture mechanics within ASTM.

During the project, all the work performed on the indirect fracture toughness estimation was re-assessed and worked into a synthesis report [5]. The MASC workshop for discussion of Master Curve applications was organised in June 2002 in Finland. A separate presentation will describe this sub-task in more details.

Prediction of loading to structures and equipment under operational and accident conditions

The loads acting on NPP components and structures may be caused by very complex incidents like thermal mixing or stratification or detonations. Thermohydraulic and thermodynamic methods have to be used to assess the temperature and pressure effects due to these loadings. The results are used as input for structural analysis.

Combination of computational fluid dynamics and structural analysis methods were tested and developed in co-operation with the TOKE project where thermohydraulic experiments were realised at the PACTEL system. The early efforts were put on the determination of the loading due to water flow and mixing by computational fluid dynamics (CFD).

Thermal stratification of hot and cold water may induce leakage in pipes of power plants [8], thus the effect of thermal loads caused by thermal stratification in a T-joint of pipes was studied. The CFD calculations [8–9] were done with the Fluent program [10], and the structural analyses were performed with the ABAQUS program [11]. The whole T-joint was simulated numerically and analysed using two stationary thermal loads solved with computational fluid dynamics [12–13]. All the loading cases are fundamentally based on the experimental tests of the TOKE project. The thermal loads determined with computational fluid dynamics were transferred to structural analyses using a transfer tool created within this work. The numerical results were in reasonable agreement with the measured ones and seemed to be realistic [13].

In a later phase, experiments of the TOKE project were modelled where non-condensable gas (air) was blown into a condensation pool. Pressure oscillations caused by sloshing were calculated for the estimation of loading transients. Tools for transferring the transient loads to ABAQUS from two-dimensional simulations with Star-CD and with three-dimensional simulations with Fluent were developed. Structural analyses including studies related to the fluid-structure interaction modelling capabilities are being performed. The pool standed in the corner of the large test room and was supported from beneath by bearers or actually a truss bracing and from the side by side stays.

In the TOKE project, experiments where steam is blown into the condensation pool are being planned. Simulations for determining loads caused by condensation water-hammer were performed with the general purpose CFD code Star-CD [14–15]. First, two-dimensional simulations for a simple NPP piping were considered as a test case. Second, two-dimensional simulations of water hammer were performed for the pool geometry of the TOKE experiment. Preliminary structural analysis of the tank is reported in [16]. Attempts to perform a three-dimensional CFD simulations of a rapid condensation of a large steam bubble are in progress. Results and remarks concerning CFD and FE analyses are reported e.g. in [17].

Accumulation of hydrogen in a BWR reactor building is possible due to leakage through the containment wall as a consequence of a severe reactor accident. Because the atmosphere in the reactor building is normal air, ignition and combustion of hydrogen can occur, leading to pressure loads on the pipes and walls. Of particular interest is whether the containment integrity can be jeopardised by an external hydrogen detonation. Structural integrity of a reinforced concrete wall and a pipe penetration under detonation conditions in a selected reactor building room of Olkiluoto BWR were studied [18].

Detonation pressure loads were calculated with the DET3D code (MOSES project). Materially non-linear structural analyses for relatively weak reinforced concrete wall and the pipe outlet were carried out with the ABAQUS/Explicit 5.8 FE code. A data transfer tool was developed in co-operation with VTT Processes and extended to handle results by the three dimensional detonation code DET3D (developed at FzK). The wall seems to resist quite well the pressure increase before detonation. The duration of the detonation is brief compared with the eigenperiods of the wall. The wall may somehow survive the detonation peak transient, but the relatively slowly decreasing static type pressure after the peak detonation damages the wall much more severely than the detonation peak itself.

Structural integrity of a pipe outlet was considered also under detonation conditions. The effect of drag forces was taken into account. According to the analyses carried out using conservative boundary conditions, the highest peak value for the maximum plastic deformation is well below the success criteria found in literature.

Fracture mechanisms of NPP structures and materials

The fundamental aim of local approach methods to fracture is to model material property dependencies arising from extrinsic and intrinsic effects. Extrinsic effects are caused e.g. by temperature, type of applied loading, specimen/structure geometry etc. Intrinsic effects are caused e.g. by ageing, crack tip constraint, mismatch, material failure micromechanisms etc. The use of local approach methods enables the determination of so called 'functional properties'. Functional properties can be either higher or lower than 'standard', laboratory, and determined properties. Use of local approach methods leads to 'best-estimates' of material and structural response, enabling identification of integrity hazards and elimination of unnecessary conservatism. Local approach methods are the state-of-the-art of 'failure micromechanism' based engineering critical analysis.

The micromechanical computational analysis methods were developed and verified for i) ductile crack propagation, ii) transition regime failure and iii) cleavage fracture. For ductile rupture, the Gurson-Tvergaard-Needleman model is considered the state-of-art method, while for cleavage the three-parameter Weibull distribution based description holds the same rank. The software basis of the research, WARP3D and WSTRESS codes developed at the University of Illinois including in-house developed routines, were taken into use and properties of the software assessed.

Numerical 2D and 3D finite element crack propagation simulations were performed with the Gurson model for three-point bending specimens with varying dimensions, e.g. [19]. Cleavage fracture was investigated similarly using the modified Beremin model, e.g. [20]. The applicability of different constraint parameters was evaluated and compared on the basis of the results. Evaluations concerning the measuring capacity of small specimens for ductile fracture resistance determination were carried out. Fracture resistance curves and the associated material parameters were calibrated. The results show that by using the applied calibration and modelling approaches the ductile to brittle failure response can be numerically simulated. Simulations for cleavage crack initiation indicated the similarities and validity of the Master curve temperature dependency. In the further development of consistent local approach methods for cleavage and ductile rupture, the emphasis is in defining viable calibration methods and in modifying the constitutive formulations to have a sound fracture micromechanical basis. A separate presentation will describe this sub-task in more details.

The investigation of failure modes of degraded pressure equipment was started as a co-operation with the AGE and METRI projects. First, the model developed in Sweden by B. Brickstad was taken under examination. It deals with the risk based design of inspections of piping where the cracks are assumed to grow by fatigue and stress corrosion. The PIFRAP program, based on this model, was implemented for test use. Within the STIN project, especially the treatment of loads and computational fracture mechanics was examined and the effect of the simplifications related to the model was estimated. The METRI project was responsible for the probabilistic aspects of the review and the AGE project of the treatment of material behaviour. As a further task, a literature survey concerning the integrity of PWR steam generator tubes was performed. The existing international practices, concerning both the non-destructive testing methods and computational assessments, as well as their applicability to VVER 440 conditions, were reviewed. The reporting of the work is in progress [21].

Validation of computational methods

The computer codes were planned to be verified by attending international comparison studies and networks. Mainly the NESC [22] network was participated. In the early phase a contribution to the final evaluation of NESC1 was made. The NESC1 tests were reanalysed with refined thermal expansion coefficient data [23]. In the NESCIII/ADIMEW project (2000–2004), a large scale experiment with a cracked bi-metallic pipe is performed. The objective of the project is to quantify the accuracy of structural integrity assessment procedures for defect-containing, dissimilar welds and quantify the benefits of advanced fracture modelling.

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, whilst advanced material models enable property transferability.

Conclusions

The results obtained in the development of methods for small specimen toughness characterisation of irradiated materials led to major improvements regarding the reliability and material usage of fracture toughness testing. At the same time the transferability of basic test results to structural integrity assessment were significantly improved.

The computational tools and methods for assessing structures under thermal and impact loading conditions were developed and verified.

Local approach fracture mechanics computation methods were found viable and worth pursuing in the predictions of material toughness as well as ductile crack propagation and brittle fracture initiation. A calibration methodology for linking the Master Curve method to local approach with Weibull statistics was identified. The methodology is dependent on determination of normalisation fracture toughness and provides a nearly identical fracture toughness temperature dependency to that of the Master Curve.

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5.2 Recent advances in Master Curve technology

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Abstract

The Master Curve (MC) methodology has evolved, from only being a brittle fracture testing and analysis procedure, to a technological tool capable of addressing many more structural integrity issues like constraint and parameter transferability. One issue that not yet has been covered by the MC is the warm pre-stress (WPS) effect. This effect, which is known to produce an effective increase in fracture toughness may be of great importance for some structural safety assessment situations, where thermal transients are involved. Here, the WPS effect is re-examined and implemented into the MC methodology.

Introduction

The MC enables a complete characterization of materials brittle fracture toughness based on only a few small size specimens. The MC method has been shown to be applicable for practically all steels with a body-centered cubic lattice structure, generally identified as ferritic steels. The method has been described in detail in several publications, and therefore, a detailed description of the method is not repeated here. It is sufficient to say that the method combines a theoretical description of the scatter, a statistical size effect and an empirically found temperature dependence of fracture toughness (Figure 16).

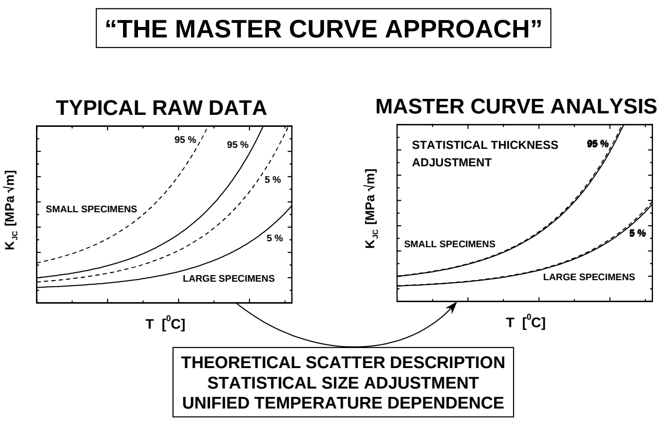

Figure 16. Principle of basic Master Curve approach.

The fracture toughness in the brittle fracture regime is thus described with only one parameter, the transition temperature T_0. The basic MC method has bee standardized in the ASTM standard E1921 [1], the first standard that accounts for the statistical specimen size effect and variability in brittle fracture toughness. Recently, the MC methodology has evolved, from only being a brittle fracture testing and analysis procedure, to a technological tool capable of addressing many more structural integrity issues like constraint [2, 3] and parameter transferability [4]. One issue that not yet has been covered by the MC is the warm pre-stress (WPS) effect.

The WPS effect describes the effect of a prior loading on the subsequent effective fracture toughness. When a crack is loaded to some fracture mechanical load level, which is lower than the fracture toughness at the temperature in question, the effect will be an effective increase in fracture toughness if the specimen is re-loaded at a lower temperature where the prior loading exceeds the fracture toughness (Figure 17). The WPS does not affect the materials fracture toughness directly. It alters the stress field around the crack and this way produces an apparent increase of toughness. The WPS effect can be connected to a variety of possible transients, some of which are depicted in Figure 17. Experimental investigations have focussed on LCF (Load Cool Fracture), LUCF (Load Unload Cool Fracture) and LPUCF (Load Partial Unload Cool Fracture). A considerable amount of research, verifying the effect has been performed during the last 30 years [5–18]. The existence of the WPS effect is unquestionable if the result is not affected by time dependent processes like strain aging. Several investigations have shown that strain aging decreases, or even removes, the WPS effect [11, 16, 17]. The WPS effect is thus not recommendable for mitigation purposes, but in a structural integrity analysis, involving a prior overload or thermal transient, the effect can well be accounted for.

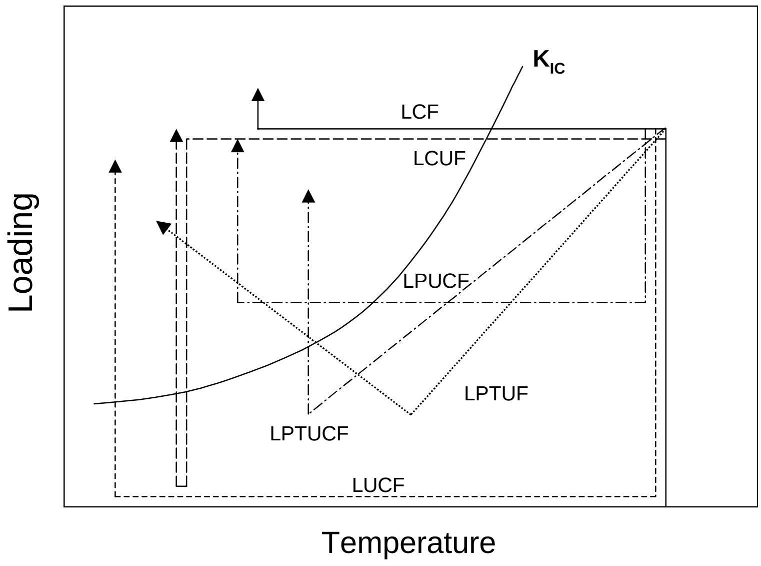

Figure 17. Principle of WPS effect.

Analysis

For the study, a total of 751 WPS test results were collected from the literature. The majority of results (456) corresponded to the load-unload-cool-fracture (LUCF) transient. The second largest group (192) corresponded to the load-cool-fracture (LCF) transient. Smaller groups corresponded to the load-partial-unload-cool-fracture (LPUCF) with 18 results, the load-cool-unload-fracture (LCUF) with 17 results, the load-partial-transient-unload-fracture (LPTUF) with 8 results and the load-partial-transient-unload-cool-fracture (LPTUCF) with 60 results.

Based on the LCF and LUCF data a new simple WPS correction was developed (Eq. 8).

$$\begin{aligned}
&K_f = 0.15 \cdot K_{IC} + \sqrt{K_{IC} \cdot \left(K_{WPS} - K_2\right)} + K_2 \\
&if \quad K_2 \geq K_{WPS} - K_{IC} \Rightarrow K_2 = K_{WPS} \\
&if \quad K_f \leq K_{IC} \Rightarrow K_f = K_{IC}
\end{aligned} \tag{8}$$

In Eq. (8) K_2 is the "unloading" WPS value. If $K_2 \geq K_{WPS} - K_{IC}$ then K_2 gets the value of K_{WPS} and if $K_f \leq K_{Ic}$ no WPS effect is present and K_f is equal to K_{IC}.

The capability of Eq. 8 to describe the WPS effect is shown in Figure 18. The new simple WPS correction was found to provide equal accuracy for all different WPS transients, making it thus generally applicable. In order to find out, if a yield stress change correction really is needed for the simple method, the LCF data, where the yield stress effect should be most straightforwardly visible, was examined further. The effect of yield strength changes during the transient, on the WPS effect, was found not to be statistically significant. Describing the yield strength effect with a power law, produced an exponent of -0.04, thus indicating a significant yield strength effect only for extremely small yield strength changes ($\sigma_f/\sigma_{WPS}<1.1$) and even then in a toughness increasing manner. The multiplication constant was taken as 0.15 as to provide a slight built-in conservatism of the correction.

Besides simplicity, the WPS correction is required also to provide a consistent lower bound failure value if a lower bound fracture toughness value is used as input. This is examined in Figure 19. It is clear that the lower bound nature of the MC fracture toughness estimates also remain after the WPS correction. The new simple WPS correction is thus capable of handling any WPS transient with satisfactory accuracy and, combined with the Master Curve, is also capable to handle the effect of WPS on the apparent fracture toughness scatter, enabling the estimation of desired lower bound values.

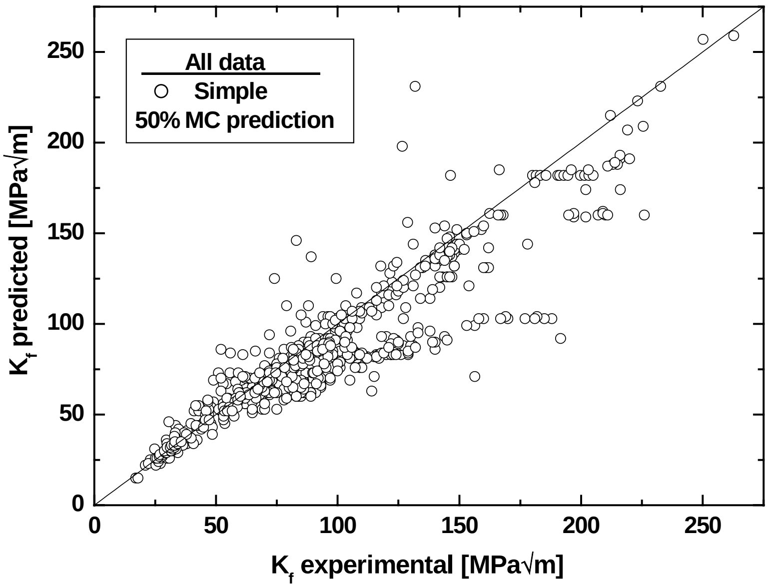

Figure 18. Predictive capability of the simple WPS correction for all 751 results using median MC fracture toughness as input.

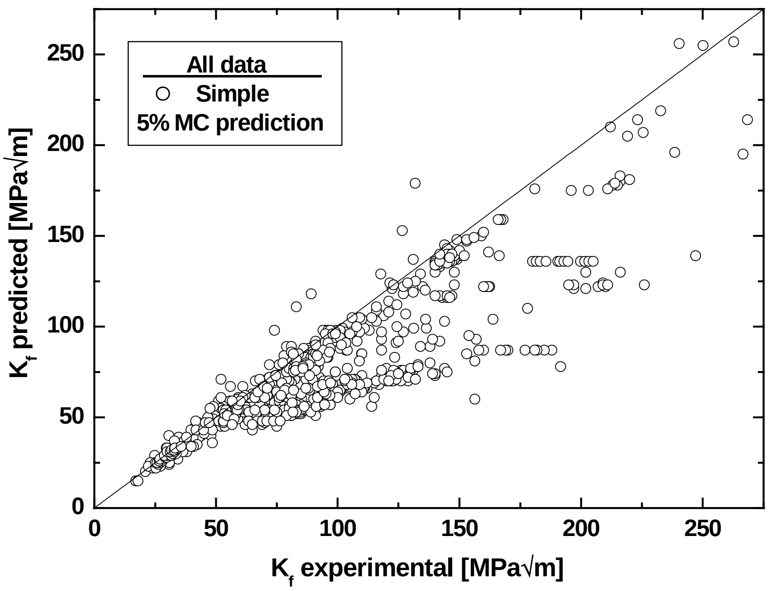

Figure 19. Predictive capability of the simple WPS correction for all 751 results using 5 % lower bound MC fracture toughness as input.

Conclusions

A comprehensive study of a large WPS data base has been performed. The data base, consisting of 751 WPS results, covered a wide variety of materials and WPS transients. In all cases studied, the existence of the WPS effect could be verified. The data indicates that yield stress changes during the WPS transient, has a negligible effect on the failure fracture toughness. A new simple WPS correction, capable of handling all transients with equal confidence, was developed and verified.

$$\begin{aligned} &K_f = 0.15 \cdot K_{IC} + \sqrt{K_{IC} \cdot \left(K_{WPS} - K_2\right)} + K_2 \\ &if \quad K_2 \geq K_{WPS} - K_{IC} \Rightarrow K_2 = K_{WPS} \\ &if \quad K_f \leq K_{IC} \Rightarrow K_f = K_{IC} \end{aligned}$$

The correction contains a slight built-in conservatism and is applicable for both best estimates and, combined with the Master Curve, lower bound type estimates to be used in structural integrity assessments. Compared to other WPS corrections, the simple correction requires less input information and is considerably easier to use.

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5.3 Local approach methods in simulating material failure

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Abstract

Local approach methods have evolved from damage mechanics models to means of providing quantified estimates of material failure behavior under various rupture mechanisms. High emphasis has been given to cleavage initiation, usually considered to induce the most profound risks to structural integrity. Overall, research has lately focused on comprehending the micromechanical background of ductile and brittle failure models and evaluation of their performance. The results have demonstrated that the promise of material parameter and failure evaluation transferability can be met, even though developments within the material models themselves are required. Current paper presents the latest results attained with respect to fracture behavior in the ductile to brittle transition and upper shelf temperature regions.

Introduction

Direct evaluation of structural integrity using local approach models introduces a tool capable of quantifying hazards posed by structural and material behavior. The use of fracture mechanics in design and failure assessment is in some practices impeded by the difficulties in quantifying the structure related constraint and transferability properties of experimental test data. It is well known that specimen size, crack depth and loading conditions may affect the materials fracture toughness. Transferability of small specimen toughness data to real structures has long been a key issue in fracture mechanics research. Methods based on the Weibull statistic, such as the "Master Curve" methodology and local approach methods of fracture have been developed and are able to characterize the scatter of fracture toughness test results and the effects of specimen dimensions on the data distribution, making it possible to define fracture toughness parameters for a given probability of failure and develop scaling models for toughness transferability.

Principles

The fundamental aim of local approach methods to fracture is to model material property dependencies arising from extrinsic and intrinsic effects. The material properties can be fracture resistance curves, ductile to brittle transition fracture toughness, the Master Curve reference temperature, failure limit strain etc. Extrinsic effects are caused e.g. by temperature, type of applied loading, specimen/structure geometry etc. Intrinsic effects are caused e.g. by aging, crack tip constraint, mismatch, material failure micromechanisms etc. The use of local approach methods enables the determination of so called 'functional properties'. These can be e.g. any of the above properties specific to a certain specimen/structure at the exhibited conditions. Functional properties can be either higher or lower than 'standard', laboratory, determined properties. Use of local approach methods leads to 'best-estimates' of material and structural response, enabling identification of integrity hazards and elimination of unnecessary conservatism. Local approach methods are the state-of-the-art of 'failure micromechanism' based engineering critical analysis. As a simplification, local approach methods take a stand in defining what effects do stress-strain-damage fields have on material and structural behavior.

Cleavage initiation

Several local approach models, in principle damage mechanics material models, exist for characterizing different fracture phenomena. For cleavage initiation, the Beremin model and its successful modifications are the commonly applied ones [1–2]. In the Beremin model the cleavage fracture cumulative failure probability is depicted to depend on the applied stress as

$$P_f = 1 - \exp\left[-\left(\frac{\sigma_w - \sigma_{th}}{\sigma_u - \sigma_{th}} \right)^m \right], \tag{9}$$

where σ_w is the Weibull stress, σ_u the scale parameter, m the shape parameter and σ_{th} the threshold stress. The Weibull stress is presented as

$$\sigma_{\mathrm{w}} = \left\{ \frac{1}{\mathrm{V}_0} \int_{\Omega} \sigma_1^{\ m} d\Omega \right\}^{\frac{1}{m}}, \tag{10}$$

where σ_1 is the first principal stress, V_0 is a reference volume set to unity and Ω is the fracture process zone. The process zone is usually defined as $\Omega : \sigma_1 \geq \lambda \cdot \sigma_0$, where λ is usually approximately 2. σ_{th} is typically identified with $K_{\min}$ from the Master Curve method [3], i.e. it corresponds to a Weibull stress at $K = K_{\min}$. The Beremin model has two parameters requiring calibration, the Weibull shape and scale parameters, and the inference is usually performed using a maximum likelihood scheme, as e.g. in [2]. Evaluation of model performance is heavily linked into the calibration procedure, since its defines the extent to which the theoretical properties of the model can be exploited.

Ductile failure

The damage mechanics model considered to show most promise for ductile tearing is the one of Gurson-Tvergaard-Needleman [4–7]. The model is basically a dilatation sensitive modification of von Mises type incremental plasticity. The flow potential is presented as

$$\Phi = \frac{\sigma_e^2}{\sigma_o^2} + 2q_1 f^D \cosh\left(\frac{q_2\sigma_{kk}}{2\sigma_0}\right) - \left(1 + q_3\left(f^D\right)^2\right), \tag{11}$$

where σ_e is the equivalent von Mises stress, σ_0 is the flow stress, f^D the damage parameter, σ_{ij} are the elements of the Cauchy stress tensor and q_i real coefficients. The damage evolution equation is given by

$$\dot{f}^D = \left(1 - f^D\right)\dot{\varepsilon}_{ij}G_{ij} + \mathrm{A}\dot{\varepsilon}_e^{pl}, \tag{12}$$

where ε_{ij} are the elements of the logarithmic strain tensor, G_{ij} the elements of the metric tensor, $\dot{\varepsilon}_e^{pl}$ is the equivalent plastic strain rate and A given in [6].

Cleavage initiation results

Cleavage evaluation has focused in unifying the local approach works to the Master Curve technology with respect to calibration procedures and at researching the behavior of the modified Beremin model.

As a result of the investigations a calibration methodology was identified capable of producing results comparable to the Master Curve and guaranteeing parameter range limited solely by the used formulation of the Beremin model. Comparison between cumulative failure probability predictions of such an approach and a specific calibrated set of parameters is given in Figure 20. Determination of the parameters relies on Master Curve temperature dependency of the normalization fracture toughness and stochastic generation of fracture toughness data using Master Curve scatter.

Local approach results have been compared to the Master Curve analyses of experimental data and good correspondence has been attained, as is presented in Figure 21.

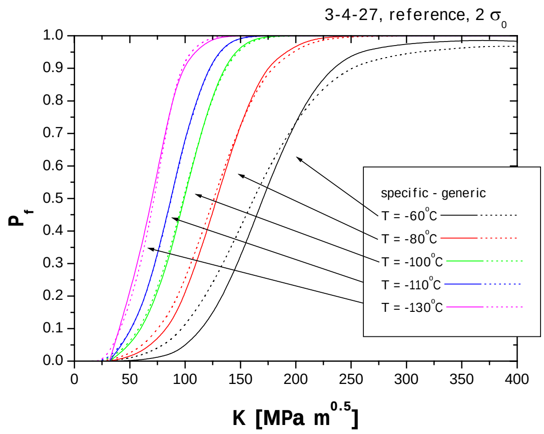

Figure 20. Comparison of generic cumulative failure probability predictions to those specific to an experimental data distribution.

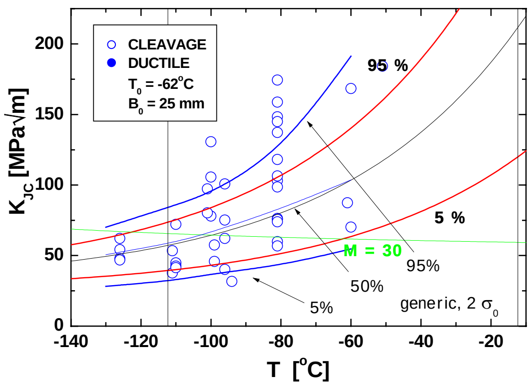

Figure 21. Comparison of local approach simulation results to those of Master Curve analysis.

Ductile failure results

Ductile crack propagation analyses have been carried out using 2D and 3D modeling for crack propagation in different size single-edge notched bend specimens. These analyses have aimed in demonstrating and improving the transferability of results of the modified Gurson model, as well as investigating the behavior of small fracture mechanics specimens with respect to topics such as measuring capacity etc.

Results pertaining the measuring capacity and behavior of small fracture mechanics specimens are presented in Figure 22. These results have been attained by calibrating fracture resistance curves of a 10-10-55 mm size specimen and predicting the behavior of other size specimens. The attained results are in harmony with experimental findings and indicate suitability of small specimens in determining ductile fracture properties. The overall performance of the modified Gurson model is depicted in Figure 23. It is demonstrated that the model and the applied modeling approach possess the means to characterize ductile crack propagation, especially if three-dimensional analyses are performed as in Figure 23.

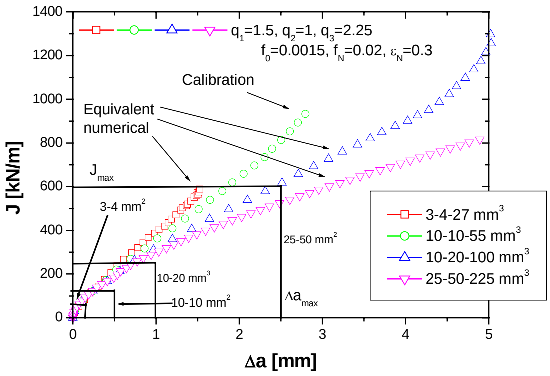

Figure 22. Predicted fracture resistance size effect for a pressure vessel steel with different size bend specimens.

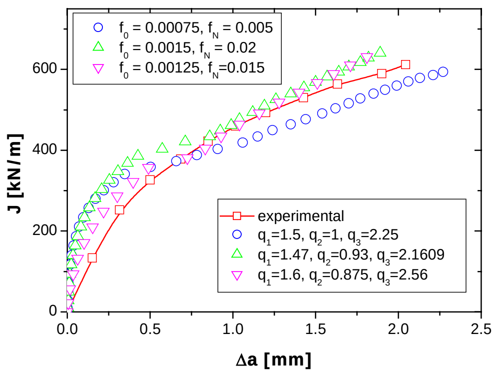

Figure 23. Comparison of experimental and numerical JR-curves for ductile crack propagation.

Conclusions

Investigation into performance of local approach models for ductile and brittle fracture was performed. Numerical analyses of fracture mechanics specimens were carried out and transferability and predictive properties of the applied material models demonstrated. The results of the work can be concluded as follows:

- The validity of small specimens for fracture resistance determination was identified using the modified Gurson-Tvergaard-Needleman model. Standard specified validity restrictions are overly conservative and the experimental indications collected over the years were independently verified by numerical crack propagation simulations.

- A calibration methodology for linking the Master Curve method to local approach with Weibull statistics was identified. The methodology is dependent on determination of normalization fracture toughness and provides a nearly identical fracture toughness temperature dependency to that of the Master Curve.

- Qualitative, and in many cases quantitative, potential of local approach models has been proven. Future work will need to investigate and improve material model transferability properties.

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6. In-service inspections and monitoring (INSMO)

6.1 INSMO summary report

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Abstract

The In-service Inspections and Monitoring project has been concentrating on the techniques and systems that are applied to examine the structural integrity of critical components by non-destructive methods. Also the monitoring of material properties has been considered to some extent. The conventional non-destructive methods are usually applied to metallic components and materials. In this project the reinforced concrete structures have been an interesting additional area. This report first lists the main objectives of the project and then concentrates on short discussion of the results of the different areas covered by project.

Introduction

In-service inspection is a very practical but also a very final link in the chain that is applied to ensure the structural integrity of the critical components of the nuclear power plants. In practice in-service inspection usually involves application of different non-destructive examination (NDE) methods. Thus, development of the NDE techniques and analysis methods is currently needed for acquiring precise and reliable information about possible flaws. Another application area is the monitoring of different material degradation phenomena where considerable resources of many laboratories are directed into national and international projects.

During the current years inspection qualification has been an important issue that has been implemented in many countries. This has required considerable efforts but it has also been considered as a methodology that will lead to more reliable application of NDE. Another significant topic has been risk-informed in-service inspection (RI-ISI) methodology. So far RI-ISI methodology has in practice meant new thinking in the selection of components and sites to be inspected. It can be foreseen that RI-ISI will produce valuable input and also requirements concerning NDE applications and on the other hand receive feed-back from the inspection results.

In-service inspections are focusing very closely on the piping and components of nuclear power plats. A special area included in the project is the structures that can be considered more or less as parts of the power plant building. These constructions are often made of reinforced concrete having own problems and requiring special inspection and monitoring applications.

Main objectives

The qualification of in-service inspection systems has been regarded as an important issue to enhance their reliability. The implementation of the qualification system has required plenty of national resources and the support of this work was one of the most important objectives in the beginning phase of the research programme. Facility to simulate and model ultrasonic inspection and to accomplish sample exercises using the available tools has been an objective that has remained through the programme. Third objective that is linked to ultrasonic method was the development of the analysis methods to gain more information from the inspection data both to analyse defect signals and to assess material properties.

The international co-operation and transfer of information from European networks and projects was considered important in the field of material property measurement using non-destructive methods. Also the risk informed in-service inspection (RI-ISI) methodology has been under development in the European network and projects and therefore the participation and follow-up of these intentions has been important.

The different structures of nuclear power plant building often made of reinforced concrete have been one special issue that clearly differs from the inspection applications of the usual metallic components. The first objective in this field was the familiarisation with the structures, possible problems and measurement needs. The second phase has targeted at different inspection or measurement trials to test the applicability of the methods.

Inspection qualification

The qualification of NDE techniques is seen as important part of the development of in-service inspections. The requirement of qualification is included in many countries to the rules regulating the in-service inspections. In Europe the methodology has been developed in the co-operation network ENIQ (European Network for Inspection Qualification). The development of the qualification system and practices has required in Finland considerable resources from different parties involved and was thus included also in the issues of the project.

Two literature studies were compiled to support the application and development of the Finnish qualification scheme. The new procedure requirements set by validation and mechanisation have been considered in a guideline paper. The other literary study covered the use of parameters that are applied in the analyses of the technical justification features. These two studies have been merged to one report that is published by the Radiation and Nuclear Safety Authority (STUK) [1].

During practical trials of the qualification, various component samples including real or artificial defects are needed. Production of these samples is expensive and time consuming phase but also the result and representativeness are of crucial importance for the qualification. Therefore production of the artificial defects and NDE signal responses measured were examined. Different notches were machined by electro-discharge machining (EDM) to fabricate reflectors that can imitate defects in ultrasonic inspection. The shape, width and depth of the notches were varied. Narrow notches up to width of ca. 0.2 mm were machined successfully. Also artificial fatigue cracks with controlled dimensions were fabricated at the edge of weld preparation.

The produced samples were measured using mechanised scanning and an automatic ultrasonic system. During the analysis phase signal amplitudes were measured and detailed B-, C- and D-scan images were produced of all reflectors. The reflector dimensions were defined following the normal sizing methods to assess the applicability of the reflectors. An example of reflector sizing is given in the Figure 24 where the lengths measured using two methods are compared with the real reflector lengths.

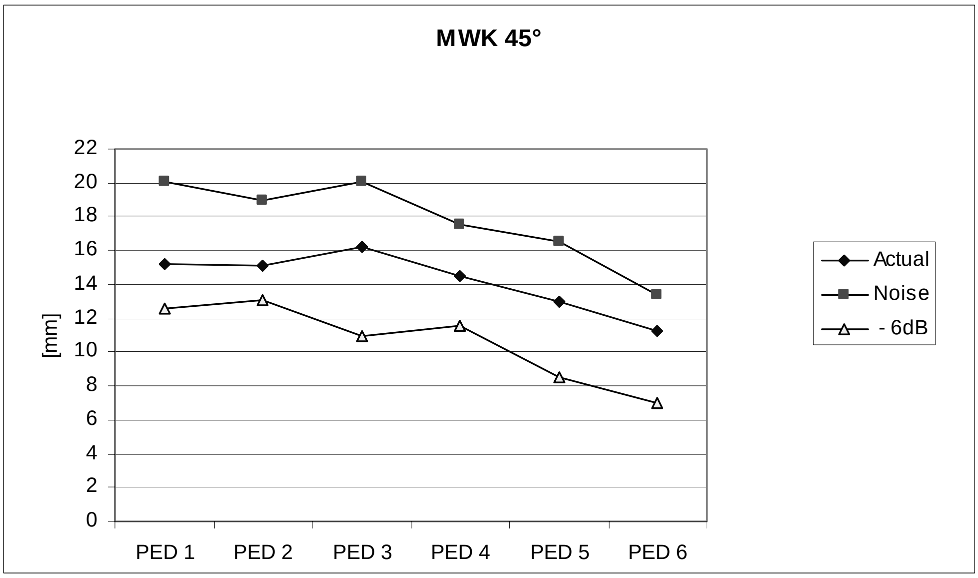

Figure 24. The results of two length sizing methods compared to actual reflector measures. Six reflectors (PED1–PED6) are included and the ends of the reflectors are defined using noise level and -6 dB drop methods.

The thorough analysis phase of the data acquired on the test reflectors offered also very useful situation for training of personnel. The results of the production of artificial defects and their ultrasonic tests are compiled in two reports [2] and [3].

Simulation of ultrasonic inspection

During the project facilities to perform simulations of ultrasonic inspection have been created. At the end of the project two computer programs with very different approaches are installed. One of these has been purchased in the project. The applications carried out in the course of the research program have offered necessary training to learn the different functions and features of the simulation programs.

The ray tracing program is a flexible tool that can be used to examine the basic requirements of ultrasonic inspections in design, test and qualification phases. This simulation program was applied for example to simulated inspection. The Figure 25 shows the ultrasonic indication patterns that correspond with the reflectors included in one of project's test plates. The same plate was inspected using real ultrasonic inspection and the results could be compared [4].

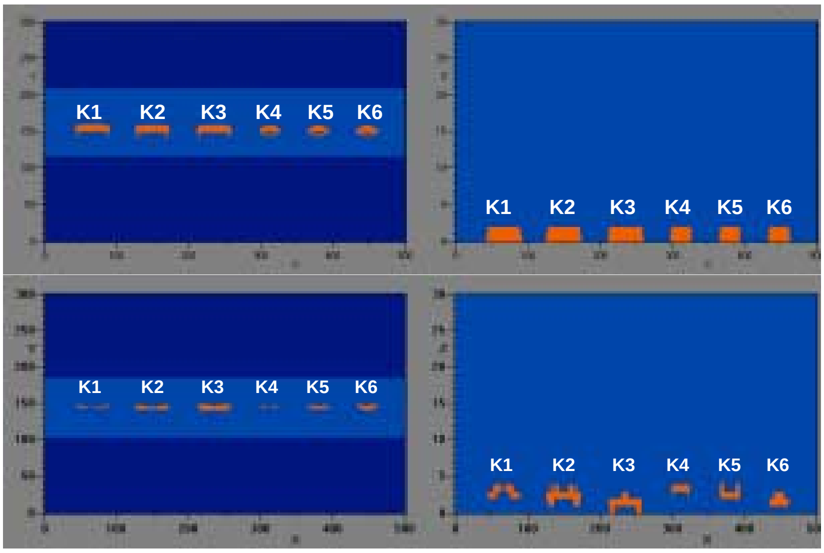

Figure 25. Indication patterns (C- and B-scans) produced using ray tracing simulation program. The reflectors of the test plate PL 1 are "scanned" from their planar side in the upper part and from the curved side in the lower part of the figure.

A very practical application of ray tracing simulation was the study of the ultrasonic probe angles for the inner radius inspection of a nozzle-to-pipe weld. In this case the inspection geometry was rather complicated and the three dimensional simulation could thus provide very valuable information. The results of the study are compiled in a report [5] and also discussed in the next section of this report.

Another simulation program applied was based on the numerical computing of wave equations. The objective of this program is to simulate the real physical phenomena of ultrasonic pulses. The current program version available is able to handle only two dimensional cases and the computing power is restricting the modelled space.

In the course of the project also a comprehensive review of the most important simulation and modelling programs in Europe was made using the literature references. The information was compiled in a report [6] where also the viewpoints of the ENIQ's guideline on the use of modelling were considered.

Risk-informed inspection (RI-ISI)

The application of risk-informed principle represents new thinking in the selection of the in-service inspection areas. Many nuclear power plants in US have already implemented this methodology in some form and this has resulted considerable changes in the inspection scope. In the course of the project the European development of this issue has been followed and information transferred by participation to the ENIQ task group on RI-ISI (TG R). The active parties of the ENIQ TG R have initiated an EU project with title "Nuclear Risk Based Inspection Methodology" (NURBIM) in that also VTT is participating.

Connections and co-operation with other projects of FINNUS has been formed to cover the broad field of RI-ISI. Contribution to the mini seminar organised by METRI project on this subject was given. Also with METRI project a joint report was produced where the structural reliability models that have already been used or are proposed for RI-ISI applications were summarised [7].

Monitoring

Through participation to the AMES/NDT network a contribution was given to a state-of-the-art report about the methods available for measuring material degradation caused by various ageing processes [8]. The members of the same network have initiated an EU-project called "Evaluation of non-destructive testing techniques for monitoring of material degradation" (GRETE). This is a round robin test project where fatigue and radiation degraded samples are tested in different laboratories. VTT is participating in the part where the fatigued samples are tested.

A small measurement case was performed on nuclear valve material. Samples with different ageing stages were available and velocities of two ultrasonic wave forms in them were measured. The results show some correlation between the impact toughness and ultrasonic pulse velocity as shown in the Figure 26. The measurements were performed using only few samples and should be completed with broader sampling.

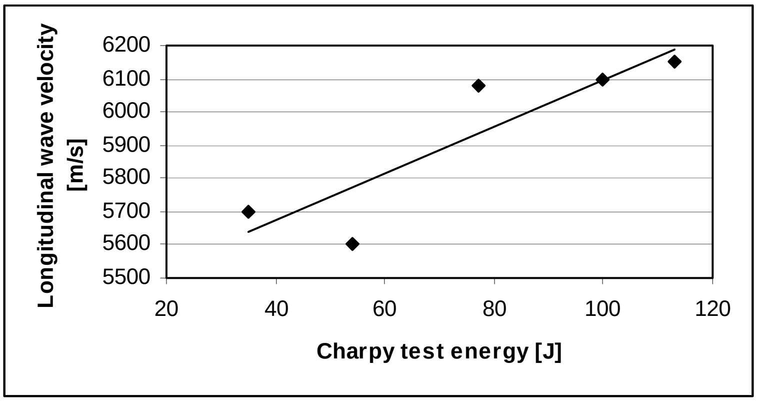

Figure 26. The correlation between velocity of longitudinal ultrasonic wave and impact toughness values measured on austenitic valve materials [9].

A new method applying spectral analysis method is developed for analysis of ultrasonic signals. The system has a data input section that is able to read acquired data of two automatic ultrasonic systems. In the analyses phase the user can define the time and frequency span to be included as shown in the Figure 27.

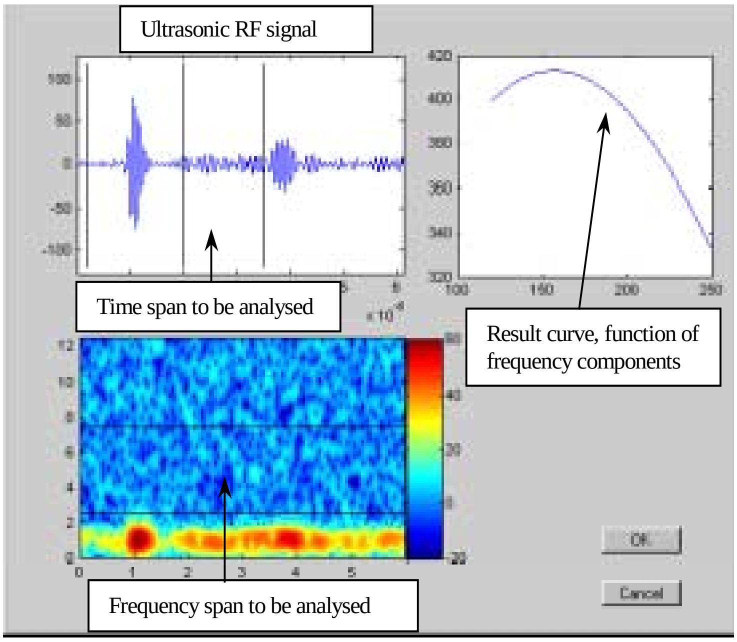

Figure 27. An example of the definition of the analyses span and result curve in the ultrasonic spectral analyses program.

The program is able to calculate the mean amplitude value of the RF-signal and the mean value of the frequency components for the chosen analyse span. The objective of this system is to provide information in the analyses phase of ultrasonic data to be used for material degradation assessment but also for interpretation of defect signals.

Reinforced concrete structures

During this subproject different structures of the nuclear power plant building have been surveyed to identify possible problem areas where non-destructive inspection or measurement methods could be applied. One important function of the project has been the familiarisation of the specialists of building technology with nuclear constructions and the co-operation of NDE-specialists from different fields.

Different measurement trials have been performed on power plant structures. For example concrete material of the turbine foundation has been monitored by ultrasonic velocity measurements to assess its state. Radiography has been applied to locate the reinforce steels in thick concrete constructions.

Applications

The establishment of the qualification has been an ongoing process during the last years and the project work contributed to this issue has been applicable in the practical tasks and problems. The fabrication techniques of test reflectors for ultrasonic inspection and especially their ultrasonic response compared to that of the real defects have been important areas when test specimens have been produced.

The ultrasonic simulation program is a suitable tool to explore quickly the basic requirements of a practical inspection. Especially when the geometry is complicated a three dimensional study can be very useful. One practical inspection case has been simulated to check the applicable inspection angles due to the request of the qualification body.

The radiographic examination of concrete walls to locate reinforce steels has already been successfully applied using conventional equipment. It has also been possible to make some conclusions about the condition of the steels. The high energy radiography system that will be available during the end of the year of 2002 is expected to penetrate also very thick walls and thus to widen the application area.

Conclusions

Methods to produce suitable artificial reflectors for the qualification samples of ultrasonic inspection have been examined. The performance of these reflectors has been measured and analysed. Also some literature studies have been performed in the area of the inspection qualification. These actions have provided support to the installation phase of the Finnish qualification system. Facilities for simulation of ultrasonic inspection have been established by installing two programs which has meant new analysis possibility. These programs have been applied to study example simulations in the project but also to some extent to practical applications.

New method based on spectral analysis of the ultrasonic signals has been designed. The first versions of the tools are programmed and some test analyses performed. The application of this method is foreseen to offer the possibilities to measure material characteristics and also additional information for defect signal interpretation.

Inspection and monitoring of reinforced concrete structures and other components belonging to the nuclear power plant building has been a new area. People with different background competence have worked in co-operation. New problems from the standpoint of the conventional NDE have been brought out. Based on the measurement trials performed it seems possible to find solutions to many of these challenges.

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6.2 Simulation examples of ultrasonic inspection

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Abstract

The paper gives examples of application of two very different computer programs that are used to simulate and model ultrasonic inspection. The real propagation path of the sound beam can be difficult to trace when the geometry is complex. Ray tracing simulation on a three dimensional model of the inspection geometry can be very useful and sometimes nearly a necessity to analyse the probe angles and scanning paths. The modelling of the sound pulse behaviour in the material and its interaction with geometric details and defects produces information about the signals that can occur during the inspection. Simulation examples show how information can be generated to support inspection design, data analyses and qualification.

Introduction

Production of test samples with real geometry and introduction of defects in the critical locations is often required when an ultrasonic inspection system and procedure are designed and their performance is tested and verified. Especially manufacturing of the samples but also the test exercises using real ultrasonic system are expensive and time consuming. Therefore different simulation and modelling tools are developed to examine inspection requirements and conditions in an effective and economical way. The simulation results can later be used also to demonstrate and verify the performance of an inspection and supporting material may be produced for a technical justification in the case of a formal qualification.

The design and performance demonstration of ultrasonic inspection can be a quite challenging task in applications having a complex geometry. One typical example is the scanning of a pipe-to-nozzle joint. When the probe is moving on the curved pipe surface around such a weld the assessment of the ultrasonic beam direction can be difficult and very laborious. The traditional scan drawings are not easy to produce except at some orthogonal cross-sections. Also the two-dimensional views are in many cases less illustrative. A tool that allows a flexible examination of the scanning situation in three-dimensional space is therefore very effective way to search and assess useful inspection parameters.

This paper presents a practical example of a case study in which ultrasonic inspection of pipe-to-nozzle geometry is studied using ray tracing simulation. This program does not model many important physical phenomena of ultrasonic testing but it provides a convenient simulation of the inspection geometry. The three-dimensional simulation tool is in this case a very practical way to check the basic requirements of the scanning.

The other example using a totally different simulation program gives an idea of the possibilities and objective when modelling of ultrasonic physical phenomena is considered. The purpose of this simulation tool is to model the creation and propagation of ultrasonic pulse in the material and also the interaction of the pulse with material variations or component geometry.

Ray tracing simulation of nozzle-to-pipe weld inspection

A typical inspection geometry of a nozzle-to-pipe weld is presented in Figure 28a. The inspection task was the ultrasonic test of the inner radius area to detect radial cracks. Because the nozzle is welded on the pipe the inner radius area can be covered most conveniently using the surface of the main pipe for scanning. Thus it is also possible to avoid the inspection through an austenitic weld.

The size of the geometrical model can be reduced by cutting out the nozzle pipe and the lower section of the main pipe as shown in Figure 28b. The scan area where the probe can move is restricted at one side by the nozzle pipe and its weld; the practical area available round the nozzle for scanning is shown with dark colour in Figure 28b. The geometry was also simplified by excluding the rounding of the inner corner of the nozzle opening. The including of the rounding increased the file size by a factor of ten in the representation suitable for transfer of the geometry into simulation program. In that case the model was too big to run in the simulation program.

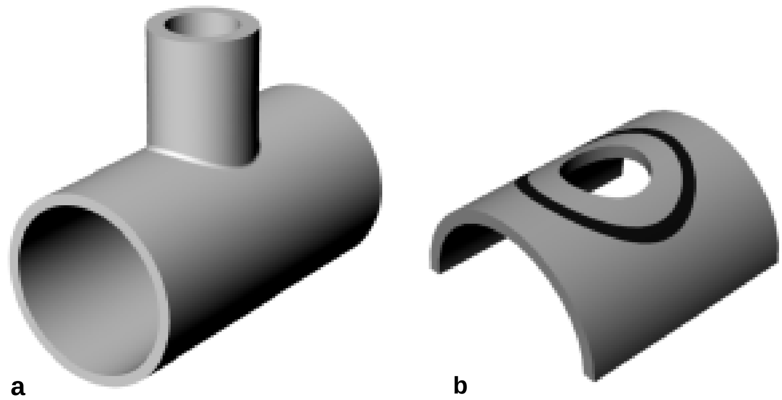

Figure 28. Component to be inspected (a). The geometric model for simulation, scanning surface round the removed nozzle pipe shown in dark colour (b).

The postulated defects in the inspection area are radial cracks starting from the corner of the nozzle opening. Three simulated defects were introduced in the model at different positions around the opening (Figure 29a). To define the inspection application it was required that the beam should be directed in an angle of 45° to the corner formed by a radial crack. This situation is illustrated in Figure 29b, where the dimensions of the component and the optimal probe location in relation to the opening and to the defect have been considered.

The main purpose of the simulation was to examine how well the ultrasonic beam could be directed into the corner formed by the crack in different inspection phases and probe positions. The reflection of the beam from the corner back to the probe was not considered as important during the simulation.

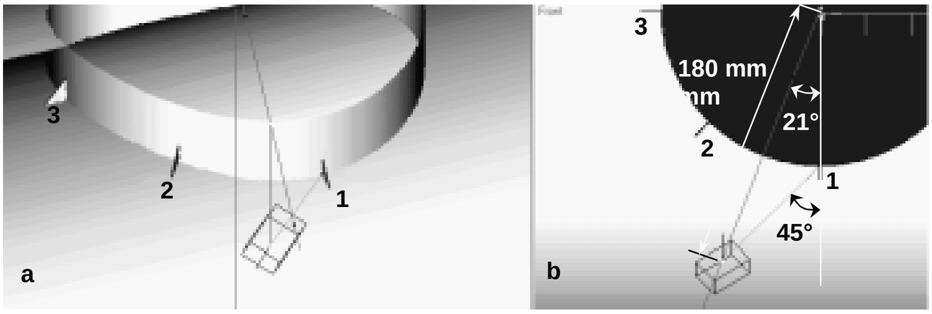

Figure 29. Three radial defects located in the inner corner of the nozzle opening in the geometric model (a). Postulated probe positioning for optimal defect detection (b).

The simulation program used in this study was "Midas Ray Tracing" (produced by Tecnatom S.A., Spain). It is based on the use of ray tracing in the geometric model of a component to be tested. The propagation direction of ultrasonic beam is shown using straight lines and it takes into account reflection and refraction according to Snell law. The ultrasonic field emerging from the probe can be modelled using a single line, a conical or a cylindrical beam shape. The capabilities and characteristics of the simulation program are described in the papers Munoz et al. [1] and Sarkimo & Pitkänen [2]. In the figures of this paper the simulated ultrasonic beam is presented by a single ray for clarity and to show the beam direction precisely.

Using simulation it was possible to find optimal inspection parameters to direct the ultrasonic beam towards the given three defects in various positions of the inspection area. The probe was positioned in a location that would allow the beam to hit the corner formed by the defect in a 45° angle (as shown in Figure 29b). Then by changing the probe angle and rotating the probe applying different skew angles the beam was directed to exactly into the corner formed by defect, opening surface and inner surface of the pipe. The optimum probe and skew angle could be read in the different windows of the program.

An existing manual inspection procedure for the inspection of the example area requires scanning on the main pipe surface round the nozzle weld and simultaneously turning the probe on both sides (± 45°) from the direction heading towards the centre of the nozzle. This scanning sequence could quite easily be simulated. Different real probe angles were chosen and their capability was checked at each flaw position. Two examples of the use of this work are shown in the Figure 30 where the points at surfaces hit by the beam are marked using dots. The defect "1" (Figure 30a) and the corner formed by it is covered reasonably well by a 65° probe. On the contrary the performance of the same probe at the defect "2" (Figure 30b) is rather inadequate.

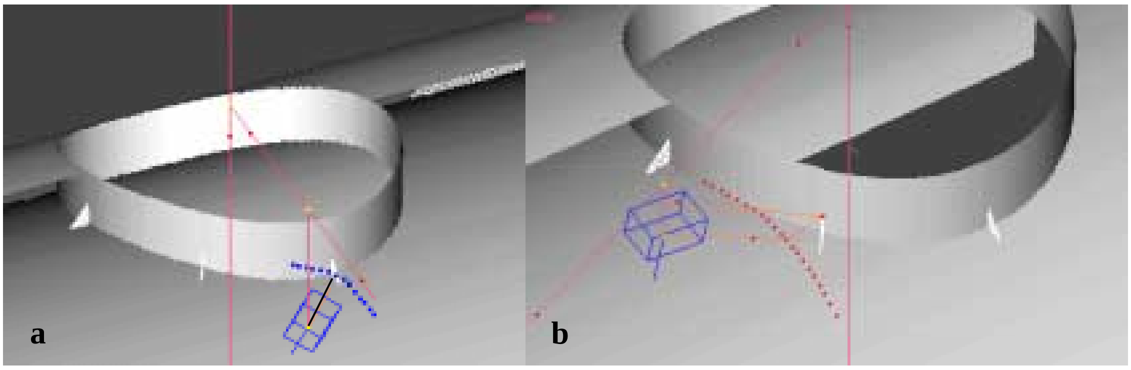

Figure 30. Inspection performance at the defect "1" (a) and "2" (b) using probe angle 65°. When rotating the probe in the case (a) in clockwise direction the beam centre line is first hitting the nozzle opening, then the corner formed by the flaw and finally the inner surface of the pipe. When rotating the probe in the case (b) the beam centre line is hitting the inner surface of the pipe clearly in the front of the defect. After rebound just the uppermost tip of the defect can be hit.

This approach could be used to examine conveniently the practical probe angles available to choose the best applicable probes for different inspection sectors round the nozzle. Also it was possible to produce evidence material for the justification of probe choice. The more details of the inspection simulation of the nozzle inner radius are presented in the paper Sarkimo and Herrero [3].

Wave propagation simulation

For the simulation of the wave propagation and interaction with reflectors was available an other simulation program called "Impulse" (Wave Process Simulation Systems Laboratory, Russia). The modelling of ultrasonic physical phenomena is based on finite difference method and numeric computation of wave equations. The paper of Alyoshin et al. [4] indicates that optimised algorithms have been sought to be able to compute the simulation on desktop computers. The current version available is capable to simulate ultrasonic tasks in two dimensional space.

This program allows the user to define many physical features of the ultrasonic probe. There are tools to design the pressure distribution created by the probe on the surface of the test object. In addition to the spatial distribution also the pressure in time domain can be defined during the design phase of the simulation. Simple drawing tools are available to design "specimens" for simulations and to define the material properties.

As an example of the simulations computed using "Impulse" program are series of snapshots presented in the Figure 31. An ultrasonic pulse is generated by a probe on the sample surface and it propagates in the angle of 45° into steel material. The plate has a notch on the back wall which traps the pulse and reflects a part of the energy back towards the probe.

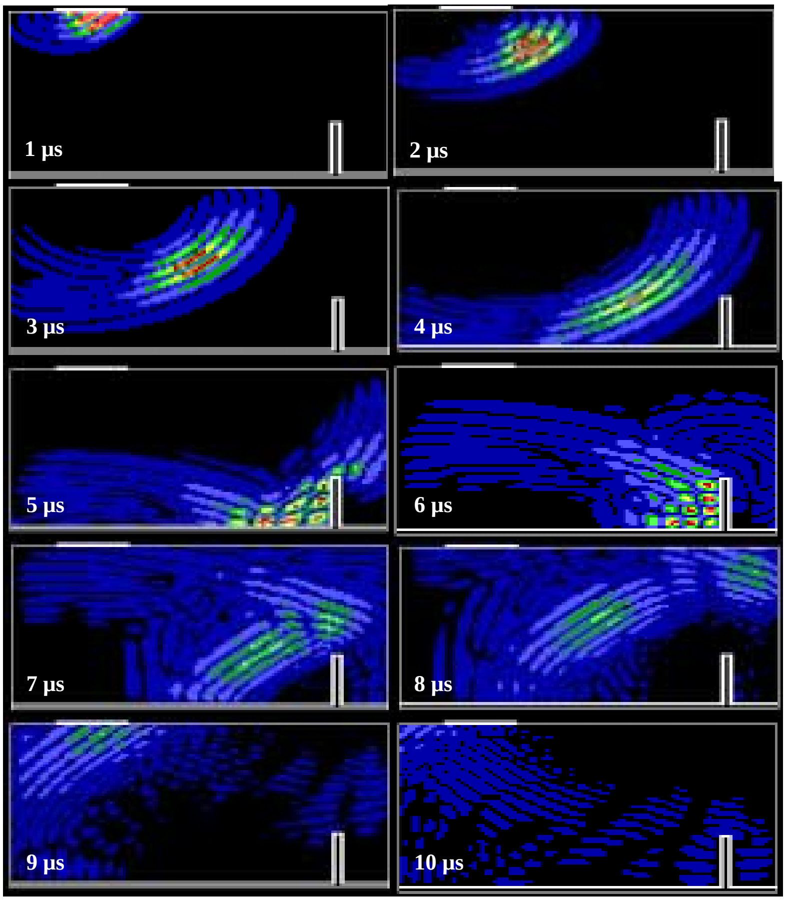

Figure 31. Simulation of an ultrasonic pulse propagation and interaction with a notch on the backside of the sample.

This kind of a simulation can be used to examine the different possible signals arising and their timing during a specific inspection task. For example one can notice in the snapshot taken at 6 µs of elapsed time that a weak annular wave is formed round the tip of the notch. In the snapshots at 6–9 µs of elapsed time a strong pulse can be seen propagating from the notch corner back to the probe (corner reflection).

Conclusions

An ultrasonic simulation program based on ray tracing can be applied for various purposes. During the planning phase of the inspection it can be used for selection and optimisation of the probe parameters. Also the generation and capability check of the scan plan can be performed by moving the probe on the inspection surface. For the qualification of inspection it is possible to produce evidence material about the coverage and other essential inspection requirements.

The benefits of the applied ray tracing program are fast applicability and possibility to work interactively with the modelled inspection. The operation based on the three-dimensional geometric models is able to visualise the inspection also in complicated applications. The possibility to import geometrical models from CAD-programs is important to have powerful and flexible resources for component modelling.

The modelling of ultrasound propagation and interaction with reflectors based on real physical phenomena is an important tool to examine how the different basic elements of an inspection affect the results. Different probe characteristics, material and geometry conditions can be simulated and studied. Also predictions of signals to be received during inspection can be produced to support the work of the analysts.

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7. Transient behaviour of high burnup fuel (KOTO)

7.1 KOTO summary report

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Abstract

There is a clear economic incentive for the industry to continue the trend of pursuing higher fuel discharge burnups, which translates into higher neutron fluences, often together with longer residence times in the reactor.

Testing and irradiation experience demonstrates that it is feasible in terms of reliability and safety to operate fuel beyond the recent burnup limits in normal conditions. Still until now, the acceptance criteria for transient and accident conditions have broadly based on early data that are often from much lower burnups, and high burnup effects must not be overlooked as regards postulated accident behaviour.

The project supports assessing the licensing criteria and evaluating the consequences of efforts for improved fuel utilisation. Statistical methods for fuel analyses have been elaborated for extensive applications. A new development featuring advanced thermal hydraulics coupled with a fuel accident performance model, the FRAPTRAN-GENFLO code, is opening an unparalleled capability for realistic transient simulation. Well-placed international participation is providing data from representative test conditions and burnup ranges relevant to model development and validation for the near future.

During the project period, the Finnish nuclear utilities were granted a discharge burnup limit extension from 40 to 45 MWd/kgU in assembly average for the fuel types in use.

Introduction

After the gradual changes that take place in the fuel during its life in the reactor core, the properties of the parts and materials are generally far from those of fresh fuel. Some of the phenomena only appear with high burnup and have become known in practice only recently. This has prompted wide research and development efforts of the industry and safety authorities for reassessing the reliability and safety of fuel in all conditions.

Through intensive research and development, fuel discharge burnups are now on levels that are up to one half higher than a quarter of century ago. In Finland the upper limit for the average discharge burnup of a fuel assembly has up to now been set at 40 MWd/kgU by the licensing authority STUK. There is still a clear economic incentive to continue, however. A calculation once made gives a total annual saving estimate of 4 million Euro in the Finnish plants, if a near-term goal of burnup increase of 5 MWd/kgU could be licensed. Of this roughly 60 % is a direct saving in fuel purchase, 40 % in spent fuel handling and disposal cost.

Testing and irradiation experience suggest that the reliability of operating fuel could be maintained beyond the burnup limits now applied in normal operation. The high burnup fuel properties that may sharply deviate from those of the fresh, however, have potentially a stronger influence in transient and accident situations. The data bases behind the criteria only partly extend to the burnup regimes now applied.

There are results, particularly from the reactivity initiated accident (RIA) test RepNa1 performed at the CABRI facility in France in 1993, together with a few similar earlier results that suggest failure of a high burnup rod at concerningly low energies. Evaluations have concluded that such behaviour is unlikely with the more recent materials, and based on a fair number of other test results the world’s regulators have refrained from posing radical restrictions in the operation. It has, however, been determined that fuel accident behaviour deserves a systematic re-evaluation. Several international efforts have been newly launched to this end, above all the OECD–IRSN CABRI Water Loop Project to study RIA. This and other international programmes are now providing experimental data from ranges of the burnups exceeding 60 MWd/kgU.

The demand of improving the situation from the Finnish viewpoint has been reflected in the objectives of the KOTO project, which is the follow-up of a succession of tasks. Apart from that, the utilities Fortum and TVO are financing the participation in the CABRI Water Loop Project, with partial support from the state through National Technology Agency Tekes, and with VTT as the main Finnish contractor. In addition, STUK is represented in the steering bodies of the CABRI project.

The changes with burnup in fuel materials include thermal, chemical, crystallographic, and neutronic phenomena. The cladding will oxidise, it will harden from irradiation and it will pick up hydrogen. The absorbed hydrogen form hydride precipitates in the metal that will make the cladding lose part of its ductility. The oxide layer will eventually grow also on the inner surface of the cladding by a reaction with UO2; forming 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 degradation of the thermal properties.

There is a specific high burnup structure that will form on the pellet. In a thin layer on the surface, the original microstructure is totally lost, the porosity and plutonium content and thus burnup and power density are high. The performance of this rim structure is not well known in accidents.

The main considerations focus on conventional accident scenarios. In RIA, sharp pulses of power and temperature concentrate on the mechanically weak rim area. In LOCA, there will always be a potential of large deformations in the cladding. When the cladding is fractured in an accident, there will be a concern of fuel relocation and dispersal and potential interaction with the coolant.

The Finnish safety authority STUK has issued a new set of regulatory guides that define the requirements on fuel performance; major changes are particularly in the guide YVL 6.2 [1]. In these guides, directions have been adopted that are in concert with the ideas that an OECD body has formulated [2]. In these, the traditional categories of design basis accidents have been replaced with a wider spectrum of events, with the probability of an event given the more restriction the more severe its consequences are. While this is physically most sound, it 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.

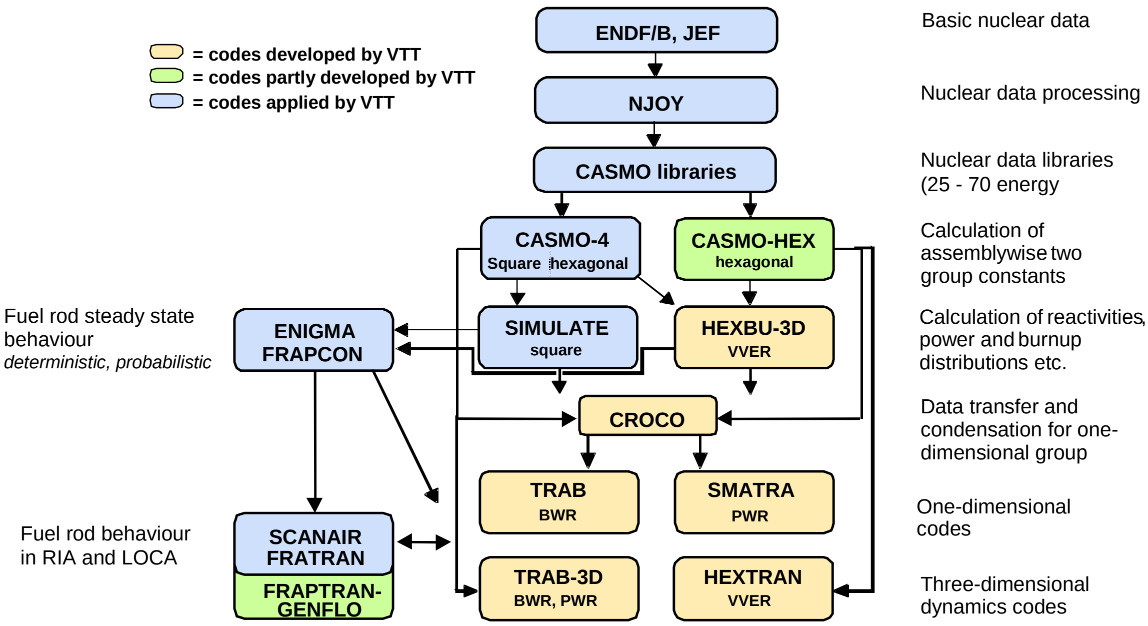
The above requirements of the reactor and fuel types in use pose a wide field of new development and applications. Research at VTT into fuel behaviour, reactor dynamics, thermal hydraulics, and material sciences will work to gradually meet this goal.

Main objectives

The main objective of the project is creating and maintaining a system of independent fuel performance evaluation tools validated up to burnup ranges that are interesting as regards the near-future needs of the safety authority and the industry in Finland. The main objective implies a continuous effort of acquiring appropriate experimental data and operating experience that are necessary for updating and calibration of the models and for validation of the integral codes by sufficient international collaboration programmes, by following the progress in general, and by participating in the work of international organisations. The project should also work as a ground for education of new expertise.

Main results

At VTT, two well established independent fuel performance code lines are now in operation. One is based on the FRAPCON-3 [3] and FRAPTRAN [4] codes due to the USNRC, for normal operation and accident conditions, respectively. The other two are the steady-state code ENIGMA [5], due to the former British Nuclear of the UK, the accident code SCANAIR [6], developed by the IRSN of France. The steady state codes also serve to generate the initial conditions for accident calculations. The system of VTT's codes for reactor analyses is visualised in Figure 32.



Developed at VTT: **HEXBU-3D, CROCO, TRAB, TRAB-3D, HEXTRAN**
Partial development by VTT: **CASMO-HEX, FRAPTRAN/GENFLO**

Figure 32. The linkage of the currently applied fuel performance codes FRAPCON, FRAPTRAN(/GENFLO), ENIGMA, and SCANAIR to the VTT's system of codes for reactor analyses.

In consent with the USNRC and together with the FINNUS/READY project, the FRAPTRAN code has been coupled to an advanced thermal hydraulic model GENFLO that has been created and successfully applied at VTT in several other contexts as well [7]. This project was responsible of the modifications in the input routines and modelling changes needed particularly in FRAPTRAN. Other changes are made to correct inaccuracies and to improve flexibility and ease of use. The combined code has been transmitted to NRC's contractor PNNL in the USA. The effort is discussed in more detail in a special report below. A systematic validation with an international peer review are foreseen in the near future.

FRAPCON-3 code has been taken into use, maintained, and tested with several cases. Subsequently, a statistical procedure has been incorporated in the code that makes it possible to yield distributions of parameters starting from distributions of input data. In the procedure there is a chance to input distributions of model parameters as well. Input features have been streamlined. The work on the NRC codes FRAPCON-3 and FRAPTRAN has been funded by STUK and the codes offer a completely independent option for safety assessments.

Of the ENIGMA code, VTT has more than ten years' experience. In previous projects the code was validated against test reactor, poolside examination, and operational experience data for VVER fuel. It was also provided with a collection of Zr1%Nb cladding material properties. Similar validation was performed against BWR fuel data. Also statistical capability had been created, lately with more applications. More recently a routine was created that will write an initial data deck for the accident code SCANAIR, with burnup-dependent features calculated by ENIGMA. Also, the model for pellet radial power distribution has been updated. Attempts to establish collaboration to produce an improved mechanistic model for fission gas release have not been fruitful mainly due to the fact that such efforts are scarce and in delicate state. The athermal enhancement of fission gas release, seen after some 40 MWd/kgU burnup has been so far described by empirically calibrated release model parameters.

The accident code SCANAIR has been acquired through active collaboration with the IRSN (former IPSN) of France. Cladding and pellet creep models have been created into SCANAIR and the solution methods have been markedly improved. This work goes within the participation in the CABRI Water Loop Project, funded by the utilities Fortum and TVO. The sponsors allow full benefits of this work for VTT and STUK. SCANAIR-3.2, the latest version of the code has been recently installed at VTT.

The two Finnish utilities have been granted an increase of the assembly burnup limit from 40 to 45 MWd/kgU for the fuel types now in use.

Two diploma theses, one on SCANAIR development [8], one on applications of ENIGMA were supervised in the project. Temporary student trainees have been employed.

Project personnel carry memberships in the OECD Halden Project Programme Group, in IAEA Working Group on Fuel Performance and Technology (TWGFPT), and OECEN/NEA Special Expert Group on Fuel Safety Margins (SEGFSM).

Related international collaboration on testing and code development

The OECD Halden project with several decades' Finnish participation remains as an essential source of basic fuel behaviour data. Annual bilateral research items emphasise the benefit from the project. VVER fuel has been irradiated in comparison with PWR fuel in instrumented rigs. Analyses on these have been made under contract from Fortum. Inclusion of high-burnup VVER fuel testing in Halden is being discussed.

Participation in the CABRI Water Loop Programme, sponsored by Fortum and TVO is expected to produce data fully representative of water reactor conditions. First two RIA experiments, in the old sodium-cooled loop, are due by the end of 2002. The water loop tests are planed to start in 2005. Project is also expected to yield high burnup mechanical material data for claddings. Loviisa VVER rod is a candidate for a test in the late phase of the programme. There is an in-kind contribution to the project in the form of SCANAIR development work done by VTT.

In agreement with the USNRC, Fortum has provided the NRC fresh samples of Russian Zr1%Nb cladding tube to be tested in NRC's comprehensive LOCA research programme at the Argonne National Laboratory. The purpose is to assess similarities or dissimilarities with the evolutionary western alloys of corresponding composition.

IAEA has launched an international Coordinated Research Programme FUMEX II (2002–2006) on fuel behaviour modelling at extended burnup with VTT participating in concultancy and calculations.

Very preliminarily, an option of inclusion of irradiated Loviisa VVER rod in the LOCA test series within the ALPS Programme of JAERI Japan has been brought about.

Applications

ENIGMA features a non-stationary heat transfer modelling and the code has been successfully applied for simulating fuel in an instability test [9]. After appropriate calibration, cladding behaviour of the rods in a Halden Project creep test series was well reproduced [10]. The statistical procedure has been applied extensively in several contracts with the utilities and STUK. From the results showing distributions of the crucial parameters like fuel maximum temperatures, fission gas release and the end-of-life pressure it is easy to judge the expected performance of the fuel and determine the fulfilment of acceptance criteria on a desired confidence level. Such evaluations have played a role in the burnup limit evaluations. A parametric effect of power is seen in an example case, Figure 33. Power reactor fuel assessments using the SCANAIR code have begun. An exaple of transient local temperature distribution is seen in Figure 34.

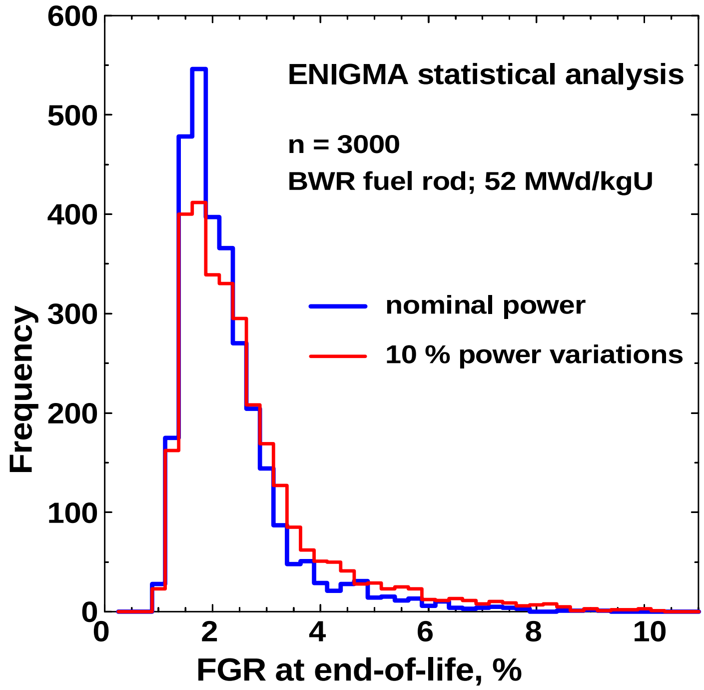


Figure 33. Application of the ENIGMA code in the statistical mode showing the parametric effect of power variation.

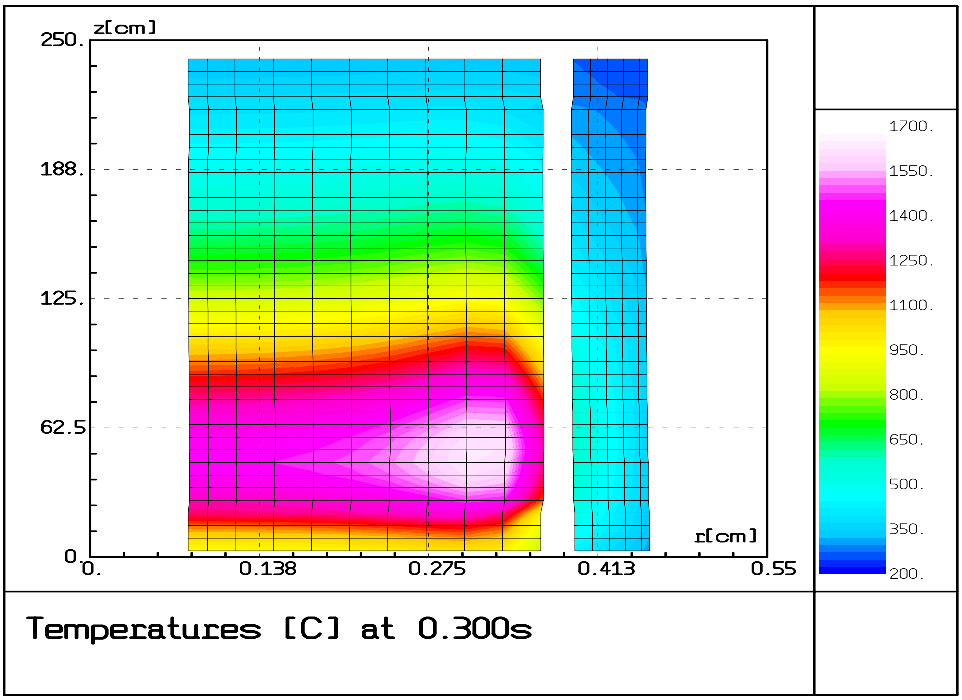


Figure 34. Localised high fuel rod temperatures in an accident; a SCANAIR result.

Conclusions

Reactor analyses have become more and more demanding for several reasons: higher discharge burnups are pursued, best-estimate type approach will be increasingly favoured, and acceptance criteria assessments are becoming more sophisticated. The purpose of this project has been to keep in pace with these developments as far as fuel performance and its modelling are concerned.

Two parallel steady-state and accident code pairs are now well established and in operation at VTT: the USNRC codes FRAPCON-3 and FRATRAN, and independently, ENIGMA and SCANAIR.

Additional high burnup data, for materials and for rod integral behaviour data are available from recently launched international programmes. The Finnish organisations are very well placed in the most essential of these, with arrangement of participation and application mainly managed by VTT.

STUK has lately granted the utilities a new assembly burnup limit of 45 MWd/kgU for the fuel types now in use. The results of this project were among those that supported both STUK and the utilities in assessing this upgrade.

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7.2 Coupling of FRAPTRAN Fuel Rod for Transient Analysis with GENFLO Thermal Hydraulic Code (KOTO & READY)

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Abstract

Reactor analyses are becoming more and more challenging. Due to pursuing higher fuel discharge burnups, fuel designs and operational conditions are subject to constant upgrading. Advances in hardware have removed many of the limitations on detail of analyses, and best-estimate type applications have become a commonplace, a practice now increasingly adopted even in safety cases. At the same time there is a regulatory trend extending the range of events the licensee is to consider. Neutronics, thermal hydraulics, and fuel behaviour are closely interlinked during a reactor transient and cannot be generally separated in a realistic description. However, attempts to combine these into coupled models have been, even at their best, impracticably heavy to use. At VTT, in consent with the USNRC, an in-house general flow model GENFLO has now been coupled with the NRC's FRAPTRAN fuel performance code. The combination takes benefits of a fast-running non-iterative thermal hydraulic model and an updated fuel performance code validated for burnups of up to 65 MWd/kgU. A code description and results of two types of analyses are given. One is a hypothetical large break loss-of-coolant accident (LBLOCA) in a VVER reactor, the other is an instability transient in a Boiling Water Reactor (BWR), for which system conditions were separately available. A systematic validation and international peer review will follow.

Introduction

Plans to include e.g. ATWS events into design basis accidents, demand of higher fuel discharge burnups, as well as more and more complicated new fuel designs set new requirements for calculation models used in safety analyses. There have been attempts to meet these with approaches from two directions: by improving the fuel models of the reactor dynamics codes, and by introducing updated high-burnup models and more advanced thermal hydraulic models in fuel behaviour codes.

Common to most fuel transient codes is the simplicity of the thermal hydraulic description of the rod-to-coolant heat transfer. This restricts the spectrum of events that can be analysed. A customary way is to use passive links from so-called hot channel analyses with system codes. This, however, is too often seen to lead to numerical instabilities and physically unrealistic results in cases with complicated power-cooling conditions. The combination of the general thermal hydraulic model GENFLO [1] developed at VTT and USNRC's fuel transient behaviour code FRAPTRAN [2] is an effort to improve this unsatisfactory situation. The combined code FRAPTRAN-GENFLO and its capabilities have been presented at international conferences [3, 4, 5].

The GENFLO module is fast running, and calculates the thermal hydraulic parameters of which local fluid temperatures and heat transfer coefficients are provided for FRAPTRAN. FRAPTRAN routines are then used to yield the temperatures of the fuel rod, and the deformation of fuel pellets and cladding including potential ballooning. The parallel way, how the coupling has been made, offers flexibility in changing the necessary data between the codes although the solution methods of the codes differ significantly.

The FRAPTRAN code

FRAPTRAN is based on the earlier FRAP-T codes, developed for the prediction of LWR fuel rods' performance during operational transients and hypothetical accidents. The phenomena modelled in FRAPTRAN include heat conduction within the fuel rod, heat transfer from cladding to coolant, elastic-plastic fuel and cladding deformation including cladding ballooning, cladding oxidation, fission gas release and fuel rod gas pressure. Burnup dependent parameters may be initialized with the FRAPCON-3 steady state code [6], which generates a link file for this purpose. The MATPRO material property package is used in these codes. Some of the models have recently been modified to reflect material behaviour at high burnup.

The input comprises of the data of the fuel rod and the definitions of the surrounding coolant channel and the transient. Essential data on the transient are the fuel rod local power history and the fuel rod boundary conditions including the system pressure. FRAPTRAN is provided with a simple thermal-hydraulic model, applicable only for homogeneous, slowly changing thermal hydraulic conditions. For other cases pre-determined boundary conditions are used. The major disadvantage of using a pre-calculation with a thermal hydraulic code is the lack of feedback. Also differences in the fuel rod modelling between the thermal hydraulic code and FRAPTRAN may cause instabilities or physically non-realistic conditions. Especially during an ATWS in BWR plants, the hot channel and the whole core may experience rapid transitions between wetted and dry states. Because of this, a dynamic exchange of detailed local data between the fuel performance and thermal hydraulic models is needed.

The GENFLO model and coupling with FRAPTRAN

The thermal hydraulic solution of GENFLO is based on the numerical solution developed for the SMABRE code [7], but its core heat transfer description has been significantly refined from that of SMABRE. In SMABRE the main interest was a reasonable accuracy of the physical models in small break LOCA and large break LOCA blowdown conditions, but for example the core heat transfer characteristics in reflooding conditions, high temperature conditions and with oscillatory flow was not emphasized. GENFLO is a fast running five-equation thermohydraulic model, where the wetted wall heat transfer, dryout and post-dryout heat transfer, and quenching models are included. The flow modes covered by GENFLO are depicted in Figure 35. The geometry described by GENFLO comprises one or several parallel fluid flow channels and an optional fuel structure. The lower and upper plena are always included but the core bypass and the downcomer may be zeroed, as is done in subchannel applications.

GENFLO solves the coolant mass, momentum and energy conservation equations, including the calculation of the axial distributions of the fluid temperature and the void fraction. As a result, the fluid temperatures and heat transfer coefficients for each axial level at each time step are supplied for FRAPTRAN, which calculates temperatures and deformation of the fuel pellets and cladding, including possible ballooning. At this stage GENFLO and FRAPTRAN use partly their own models, i.e., for fuel and cladding temperatures, cladding oxidation and hydrogen generation. The aim in the future is that the fuel specific calculations are made for both codes by FRAPTRAN and the coolant specific calculations for both codes by GENFLO.

In the coupled code, FRAPTRAN is the master code calling GENFLO, which provides the thermal hydraulic conditions for the whole channel. This calculation is performed only once for each time step, even if a number of iterations is done in FRAPTRAN during the time step. In the beginning, GENFLO is required to make a steady state calculation before any coupled code calculation. In the coupled code calculation, FRAPTRAN dictates the time step length, typically 0.01–0.05 s, but the calculation is fast because GENFLO is non-iterative and effective numerical methods are applied.

The system behaviour and boundary conditions needed for a detailed core simulation may be calculated with various system codes such as RELAP5 or others. At VTT, also the three-dimensional BWR or PWR reactor dynamics codes TRAB-3D and the simulator APROS have been used. The data exchange between a system code, GENFLO, and FRAPTRAN is illustrated in Figure 36. At present, the boundary conditions provided for GENFLO from the system code are the mass flow and the enthalpy at the channel inlet, the pressure at the top of the channel, as well as the total power and the power profile of the fuel rod.

Although the GENFLO model has been tested in different surroundings, the coupled FRAPTRAN-GENFLO code will still need verification, including comparisons with the existing thermal hydraulic models in FRAPTRAN.

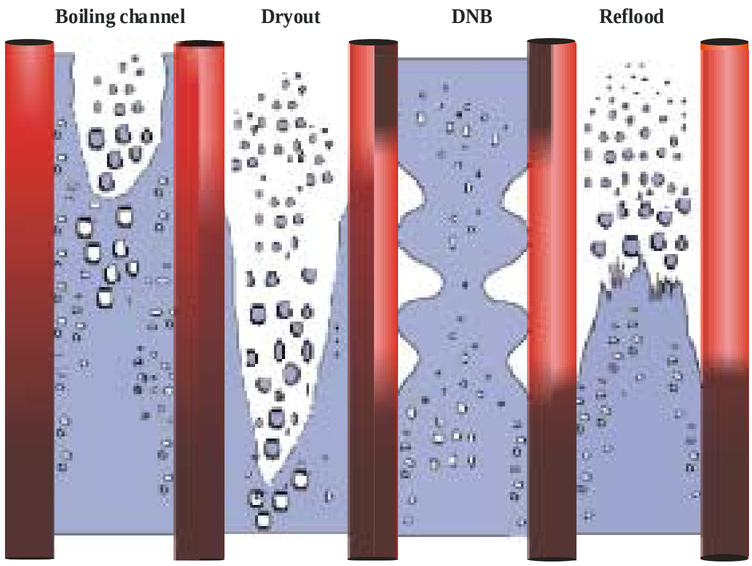

Figure 35. The channel flow modes in GENFLO.

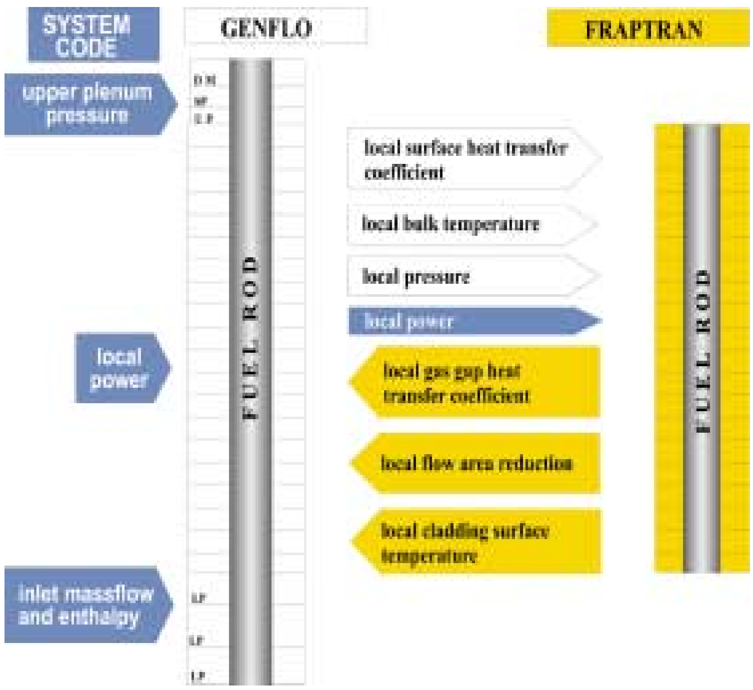

Figure 36. Data exchange between the system code, GENFLO and FRAPTRAN.

Two example calculations using FRAPTRAN-GENFLO

In the first example FRAPTRAN-GENFLO is used to analyse fuel behaviour in a hypothetical large break LOCA at the Loviisa VVER-440 power plant. The engineering simulator APROS has been applied for the system behaviour in the LBLOCA. The initiating event for the transient is a double-ended cold leg break. The transient is assumed to occur at the beginning-of-life with no significant fuel burnup. As a comparison, the same case was calculated with FRAPTRAN using the simple built-in coolant model.

At the beginning of the transient the fuel temperature quickly drops because of the power decrease. Later, due to the loss of coolant inventory, the fuel temperature starts to increase. The coupled code predicts that even a small amount of water in the channel inlet from the low pressure safety injection is sufficient to first stop the temperature increase and then to temporarily decrease the temperatures. The fuel centreline temperatures for all axial nodes from the FRAPTRAN-GENFLO calculation are presented in Figure 38. The difference between the cladding and fuel temperatures is small after about 50 s. FRAPTRAN-GENFLO predicts no ballooning or failure of the rod, but FRAPTRAN separately, using the unrealistic built-in simple coolant model, predicts higher temperatures, ballooning and a failure within 300 s.

The second example is an instability transient during an ATWS in a BWR plant. The case is based on a real oscillation incident in a BWR reactor during the reactor start-up. The incident was safely terminated by the normal operation of the reactor safety systems. To test the performance of the new model combination, the case was hypothetically extended assuming no actions of the safety system. The transient was recalculated with TRAB-3D as an ATWS case. The oscillations of boundary conditions were artificially continued in time and amplified. The boundary condition of the channel inlet mass flow is shown in Figure 37. The flow rate oscillation finally leads to temporary flow reversals (negative mass flow). The fuel was assumed to have high burnup, and the FRAPTRAN calculation was initiated by the FRAPCON-3 link. The calculated fuel average burnup was 62.3 MWd/kgU.

The results of the FRAPTRAN-GENFLO calculations for this BWR instability case show that with a power cycle period of about 2 s, the fuel rod remains covered with water until the time of flow reversal, and only local or temporary dryout or DNB conditions may occur. The flow reversal soon leads to high cladding temperatures at the upper part of the fuel rod, but the highest fuel temperatures later occur at the lower part of the rod, where the linear power is higher. Figure 39 shows the temperatures of the rod (in degrees K) as calculated with FRAPTRAN. The axial node 1 is at the bottom; the radial node 1 is at the pellet centre, node 17 at the cladding surface. At the end of the calculation the cladding is quite soft and in contact with the fuel pellet. The plenum gas pressure remains below the fluid channel pressure during the whole transient.

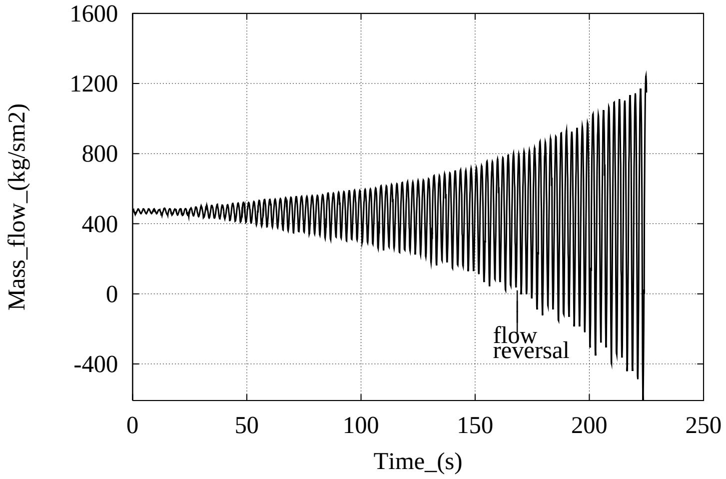

Figure 37. The mass flow rate boundary condition at channel inlet in hypothetical BWR instability case.

FRAPTRAN-GENFLO proves to be a proper tool for studying oscillation phenomena in a single subchannel, although continuation of the oscillation in real geometry may be much different from the oscillations assumed in this example. The results further suggest that the critical heat flux correlations in GENFLO need a review.

Conclusions

A thermal hydraulic subchannel model GENFLO (=GENeral FLOw) developed at VTT has now been coupled with the USNRC fuel transient code FRAPTRAN. The calculation of the example cases, Loviisa Large Break LOCA and a BWR instability case, with the coupled code has shown that GENFLO can be successfully used as a hydraulic model in the FRAPTRAN code. Due to the non-iterative solution techniques and advanced numerical methods applied in GENFLO, the combined code can demonstrate high performance with short running times.

The parallel coupling of the codes makes it easy to make changes and track errors in the FRAPTRAN and GENFLO codes, and further features may be added easily. Several improvements in the codes were already introduced during the course of the work.

In the example case 1, no cladding ballooning was predicted with the coupled code. The separate FRAPTRAN calculation would suggest ballooning, cladding deformation and rod failure with these partly unrealistic boundary conditions. Calculation of the example case 2, the BWR instability case, shows that with a small cycle time the channel is wetted and no sudden rod failures occur before flow reversal. The pellet-to-clad gap is expected to be closed at the end of the calculation. For more definitive analyses of both cases the input and boundary conditions ought to be studied and repared more carefully. The results of the first version of this coupled code are most encouraging and warrant further development. Careful validation against experimental data and a peer review should follow.

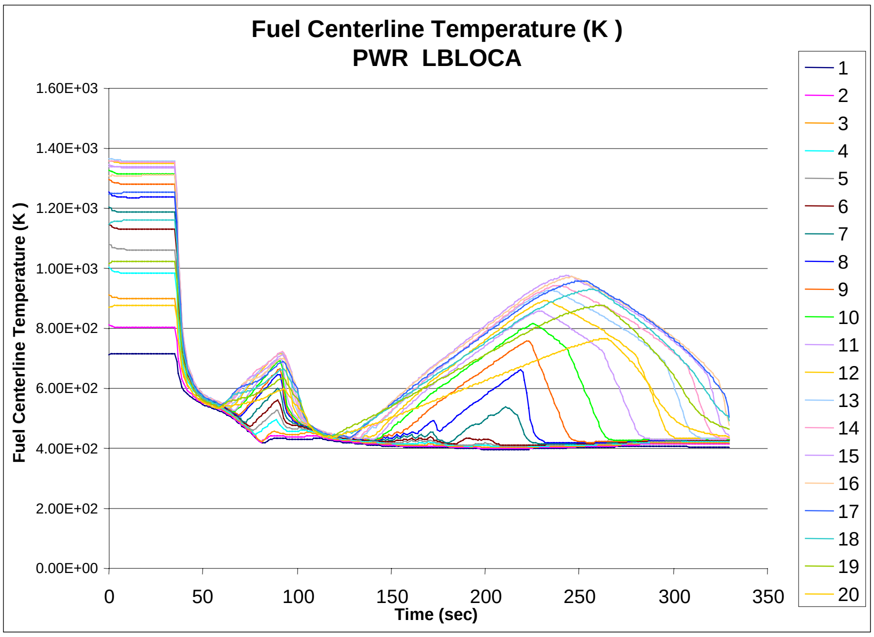

Figure 38. Fuel centreline temperature calculated using FRAPTRAN-GENFLO.

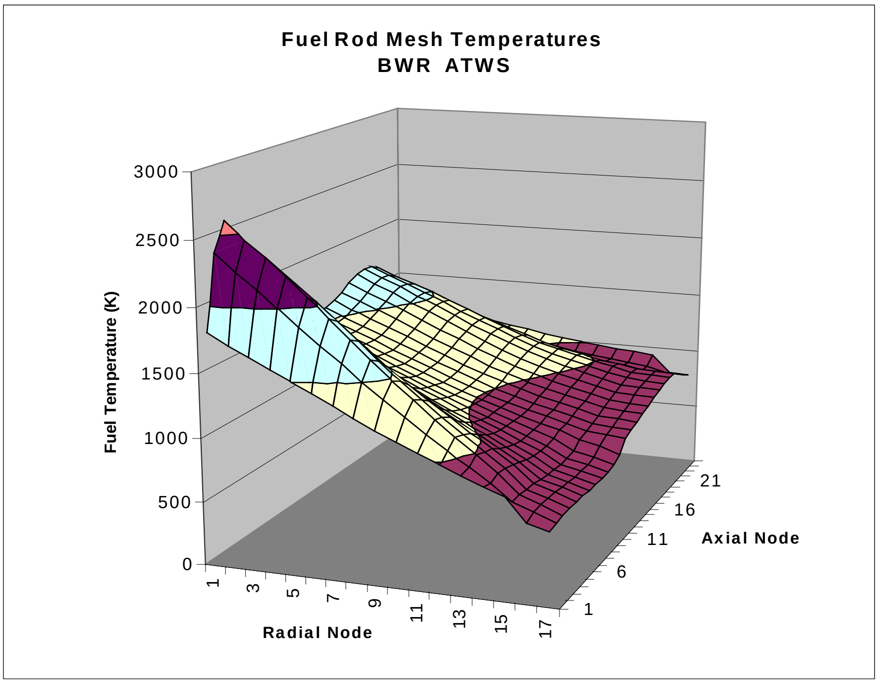

Figure 39. Fuel rod temperatures calculated using FRAPTRAN-GENFLO.

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8. Reactor physics and dynamics (READY)

8.1 READY summary report

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Abstract

As a result of upgrading, adapting, developing and validating several reactor physics and dynamics codes of VTT's code system more accurate calculations and predictions of physical phenomena can now be performed. Three-dimensional reactor physics codes have been introduced and validated for various applications, especially for out-of-core flux, criticality safety, and dose rate calculations. A multi-temperature MCNP cross section and scattering law library has been created. Most recently Monte Carlo technique has been applied in burnup calculations. Development of a new advanced nodal model has resulted in a highly promising new BWR simulator code, ensuring independent calculational capabilities for steady state and safety analyses. Validation of VTT's three-dimensional BWR dynamics code has been completed, and analyses with three-dimensional core models can be carried out for PWR, BWR and VVER type reactors. Until now, the transient analyses for BWRs have been done with one-dimensional models. Fuel models of the dynamics codes have been improved, and USNRC's fuel behaviour code has been coupled with VTT's advanced hydraulics model. The coupled code has been delivered to USA in agreement with USNRC. Development and application of a sophisticated hydraulics solver in reactor dynamics has resulted in a BWR circuit model that has been tested in steady state calculations.

Introduction

VTT Processes has created and maintained 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 reactor physics the main objective was to ensure reliable and continuous use of the extensive code system both by updating and validating the codes and the ways to use them, and training of new personnel. Special emphasis was in application of Monte Carlo technique, applying the nuclear data processing system NJOY'97 and validating three-dimensional out-of-core calculation system against experimental benchmarks. In order to be able to model cores with increased heterogeneity, an advanced nodal method (including pin power reconstruction) was to be developed. The reactor physics code system largely relies on internationally widespread computer codes acquired through the NEA Data Bank. Therefore, it has been 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 was planned. Fruitful co-operation on VVER safety was also to be continued within AER.

In reactor dynamics the objectives were to complement and validate the calculation system for complex scenarios, such as ATWS, boron dilution and BWR core stability. The three-dimensional reactor dynamics codes TRAB-3D and HEXTRAN were to be further validated against plant measurements and international benchmarks. Co-operation with the Behaviour of High Burnup Fuel in Accidents project (KOTO) included 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 were to be improved by taking into use the numerically accurate solution method PLIM. The CFDPLIM solver itself was also to be further developed to have a new more robust version for the coupled calculations. Opportunities of the six-equation model SFAV in modelling thermal hydraulic phenomena were to be studied.

Validation of three-dimensional reactor dynamics codes

The three-dimensional reactor dynamics code TRAB-3D [1] developed at VTT has been validated for coupled neutronics and thermal-hydraulics analyses of PWRs and BWRs in square lattice geometry [2, 3]. In addition to the core dynamics TRAB-3D includes BWR circuit models; the combination has been succesfully validated by calculating real TVO plant transients [4]. TRAB-3D has also been validated by calculation of the OECD/NEA Main Steam Line Break benchmark [5]. 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). Recently the calculation of the separate core exercise of OECD/NEA BWR turbine trip benchmark has been completed with TRAB-3D.

The load rejection test carried out at Olkiluoto 1 in 1998 in connection with the power uprating was chosen for validation of TRAB-3D, because data on local power measurements was available for this test, making the test a good case for validating three-dimensional core models. The test includes a partial scram, which is asymmetric, thus a full 3D model of the core was needed. The transient was successfully calculated with TRAB-3D. Preliminary results indicate that code predictions of local power during the test are in very good agreement with measured local values. In earlier cases, only steady state power distributions have been compared to measurements or results from other computer codes. See also a special report below.

Concluding from the validation work, the TRAB-3D code is ready for licencing applications. Until now, the transient analyses for BWRs have been done with one-dimensional models. A new methodology, needed in proceeding to the more realistic three-dimensional analyses, is being studied as a Nordic co-operation effort within the NKS research programme framework

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) has been developed at VTT [6]. The AFEN 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 e.g. improve the accuracy of the ex-core detector flux estimates. Avoiding difficulties of earlier methods, the solution method of AFEN enables straightforward pin power reconstruction. The model has been succesfully tested against IAEA-2D benchmark and with a simplified core of a BWR in cold state against Studsvik's SIMULATE-3 code.

In order to create a new, sophisticated BWR simulator code ARES (AFEN Reactor simulator) new modules were developed for thermal hydraulics, cross section and burnup calculation. The thermal hydraulics module is a four-equation based stationary-state thermal hydraulics model. The new cross section model is a hybrid between table interpolation and polynomial fitting, using polynomial fits with respect to the time-critical momentary quantities. The ARES model has been tested at full power BWR conditions, but a more thorough testing is planned after the burnup module has been completed. See also a special report below.

HEXTRAN is a three-dimensional hexagonal reactor dynamics code developed at VTT [7]. 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 [8, 9]. Figure 40 shows the cross section of the reactor core and the different material regions with the three locations of ionisation chambers in its surroundings. The shaded region (excluding the centermost assembly) in the core indicates the fuel assemblies for which the kernels are calculated. The comparisons against Loviisa and Czech Mochovce rod drop experiments showed increased consistency (20→10 % difference). Later HEXTRAN was provided with a new description of control rods with full albedo matrix consisting of partial albedos from face to face of the hexagonal region [10]. This further improved the agreement with measurements giving a relative deviation of -7...-1 % for Loviisa and of -4...-2 % for Mochovce.

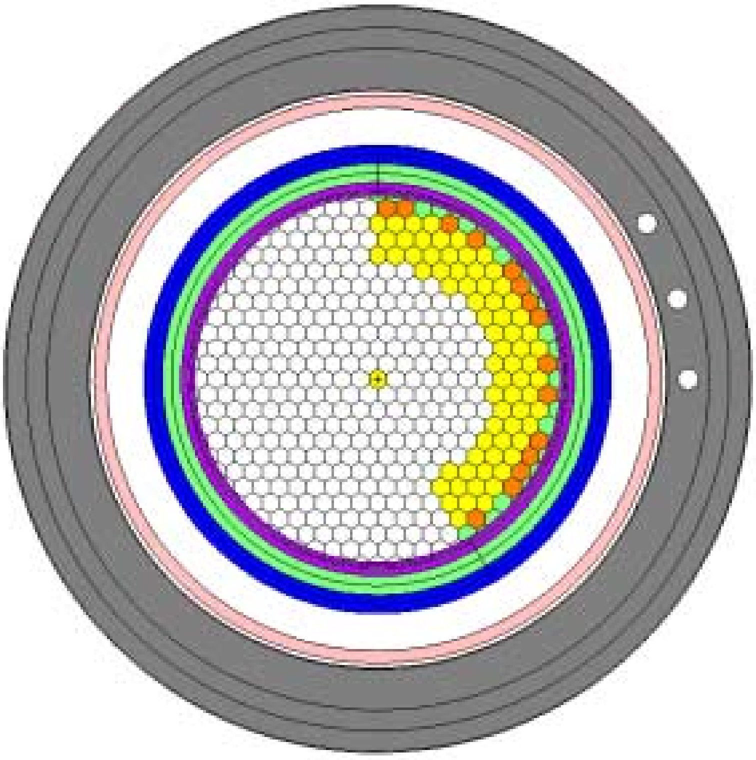

Figure 40. Horizontal cross section of the reactor in calculation of detector response kernels.

In dynamic analyses the thermal hydraulic conditions within the reactor core may have a large variation, which sets a special requirement on the modeling of cross sections. The accuracy of the present cross section model is sufficient for the analysis of most transients, especially when the data is tailored to cover the transient conditions. In real analyses the thermal hydraulic conditions in creation of the cross sections have been carefully selected for best estimate results. For transients with large variations the present model is known to be insufficient and less experienced analysts would also benefit from the opportunity to use only one wide-range set of data in all transient conditions. Therefore a new, wide-range model of cross sections developed at VTT and Fortum Nuclear Services for HEXBU-3D/MOD6 has been included as an option into HEXTRAN. In this model the nodal cross sections are constructed from six state variables with a polynomial of more than 40 terms. Coefficients of the polynomial are created by a least squares fitting to results of a large number of calculations with a fuel assembly burnup program. Depending on the choice of state variables for the spectrum calculations the cross section model is capable to cover local conditions from cold zero power to boiling full power state of a fuel assembly. Further testing and comparison between the old and new models in various VVER benchmarks and transients will still be needed

A pin power reconstruction model developed for TRAB-3D has been completed [11, 12]. The model is applied as a stand-alone program after a full TRAB-3D calculation, using nodal data written by the main code. Node-wise power distributions, corner flux discontinuity factors and the control rod history parameters are interpolated from a matrix of CASMO-results.

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 behaviour codes [13].

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 [14]. 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. Most of the models have also been implemented in the three-dimensional TRAB-3D code, together with a new model allowing interpolation for gas gap conductance from a table with dependencies on pellet burnup and temperature.

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. The fuel behavior model of FRAPTRAN has recently been upgraded but its hydraulic model is still based on the old RELAP homogeneous model. As a co-operation between the READY and KOTO projects at VTT, FRAPTRAN has been coupled with a more sophisticated hydraulic code called GENFLO (GENeral FLOw) [15] that has been developed at VTT [16, 17, 18]. The GENFLO code is also coupled with separate neutronics model in RECRIT code for BWR recriticality and with several models in Loviisa severe accident training simulator APROS-SA [19]. GENFLO is based on fast, non-iterative five-equation model [20] including also rewetting capabilities up to the melting temperatures of the fuel rods. In agreement with US Nuclear Regulatory Commission the coupled code FRAPTRAN-GENFLO has been delivered to Pacific Northwest National Laboratory in USA, but its testing and validation will continue also at VTT. See also the KOTO project special report above.

Advanced thermal hydraulics modelling

Piecewise Linear Interpolation Method (PLIM) [21] 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 e.g. 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 CFDPLIM solver has been modified to include diffusion type second order differential terms for turbulent mixing and to be numerically more stable in reactor dynamics applications. To facilitate the treatment of these terms a new iteration step solving the equations simultaneously in every point of a flow channel is being added to CFDPLIM. A separate model for the BWR water level has been developed to allow description of relevant phenomena in the present BWR dynamics model.

The BWR circuit model for TRAB-PLIM, including the downcomer, pumps, lower plenum, core channel, by-pass channel and a riser part above the core, together with the original TRAB upper plenum and steam line models, has been completed. Steady-state calculations and comparisons with TRAB results have been performed [22]. The TRAB-PLIM correlation selection has been expanded to correspond to that of the production version of TRAB. 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) [23, 24] 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. The amplitude growth driven by forced oscillations was then observed in experiments planned by STUK and made in the test facility of Fortum Nuclear Services. The safety significance of the observed phase separation remains still unclear.

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, 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 studied. The validation has been continued with the calculation of the VVER Fuel Burnup Credit Benchmark [25]. The agreement with the solutions provided by other participants in the benchmark was good. The calculations confirmed that axial homogenization of spent fuel is non-conservative at high burnups. The Monte Carlo method has also been applied to study the VVER-440 control element efficiency and the Tokaimura criticality accident in 1999.

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. At VTT NJOY has been applied to generate multitemperature neutron cross section and thermal scattering data for the MCNP Monte Carlo particle transport code from the ENDF/B-VI revision 5 evaluated data library [26]. The new data enables wider use of accurate Monte Carlo analyses. It was utilized internationally in a EU project evaluating the feasibility of a reactor working in supercritical pressure conditions with high efficiency (HPLWR). [27]

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. Previously, the two-dimensional DORT discrete ordinate transport code has been used at VTT for out-of-core flux calculation. Now, the three-dimensional TORT discrete ordinate transport code has been validated against the VENUS-3 benchmark [28, 29]. A discrete mesh generator program DIMER [30] has been developed at VTT to automate mesh, material distribution and source distribution generation for TORT calculations. The benchmark was studied in both Cartesian and cylindrical geometrical models, to accommodate for the geometric features of the square lattice fuel region and the cylindrical outer regions (reflector, barrel) with different approximations. Figure 41 shows an example of the response distribution plot above the core midplane of the cylindrical case. The calculated benchmark results were in good agreement with the reference results in both geometries.

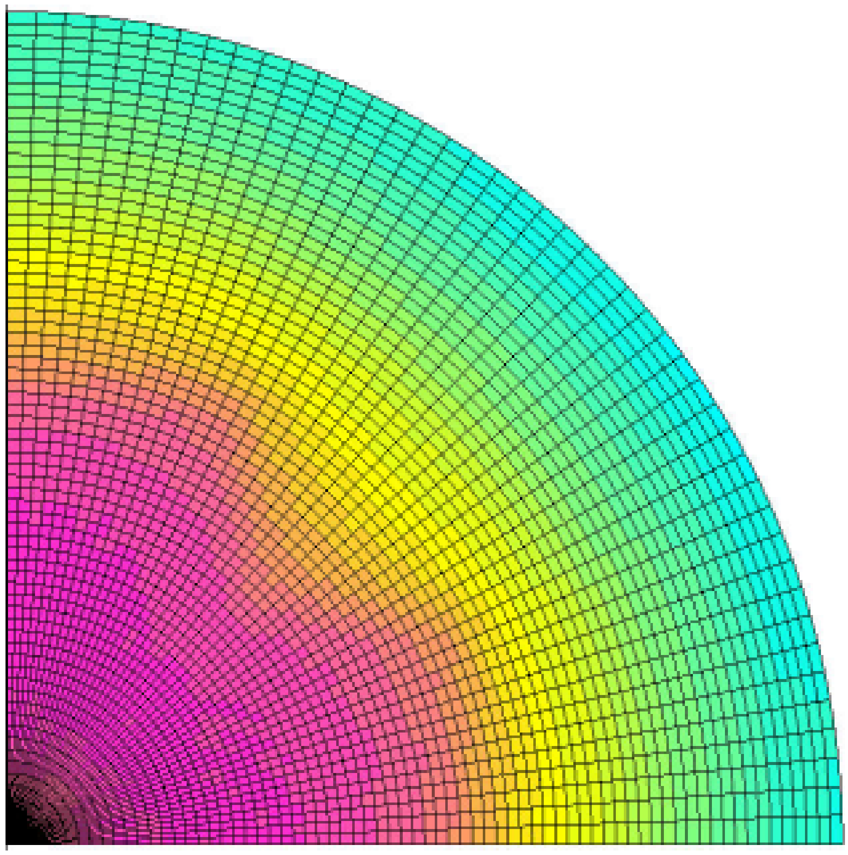

Figure 41. A result of VTT's VENUS-3 benchmark calculation with TORT: the Ni-58(n,p) response distribution above the core midplane in the cylindrical case, logarithmic scale.

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) [31]. Also criticality calculation codes have been validated within AER co-operation [32]. 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) [33, 34]. Recently, international interest in three-dimensional dynamics calculations and availability of new models has resulted in a Nordic NKS-project "3D transient methodology for the safety analyses of BWRs", co-ordinated by VTT.

Education of experts

One of the goals of the READY project is to maintain the reactor physics and dynamics expertise in Finland. The project supports both undergraduate and post-graduate studies. Four master's theses and one licentiate thesis have been completed in 1999–2002 [6, 22, 29, 35, 36]. The project employs presently six young persons (YG): five post-graduate scientists and one research trainee. The YG has participated in international training courses and summer schools.

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 fuel management work for the Olkiluoto-1 BWR, utilizing also the expert system CORFU [37]. The ICFM of the Loviisa VVERs is made with VTT’s HEXBU code, 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-HEX and CASMO-4 codes.

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 simplistic 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, and fusion studies. The expert knowledge has also been applied to international contract research and EU projects, as well as to improve safety of VVERs in Central and Eastern Europe, eg. within Phare and Tacis projects [38, 39].

Conclusions

More accurate calculations and predictions of physical phenomena have become possible as a result of upgrading, adapting, developing and validating several reactor physics and dynamics codes of VTT's code system.

More accurate, three-dimensional reactor physics codes, like the MCNP Monte Carlo code and the TORT discrete ordinates code, have been introduced and validated for various applications, especially for out-of-core flux, criticality safety, and dose rate calculations. A multi-temperature MCNP cross section and scattering law library has been created with the NJOY data processing code. Most recently Monte Carlo technique has been applied in burnup calculations. Continuous research efforts are required to improve understanding of the theoretical basis of the codes and nuclear data libraries.

Development of a new advanced nodal model has resulted in a highly promising new BWR simulator code, ensuring independent calculational capabilities for steady state and safety analyses. The development and validation of the simulator will be continued.

Dynamics analyses with VTT's validated three-dimensional core models are now possible for both VVER (PWR) and BWR reactors. Until now, the transient analyses for BWRs have been done with one-dimensional models. A new methodology, needed in proceeding to more realistic three-dimensional analyses, is being studied as a Nordic co-operation effort within the NKS research programme.

The new coupled code FRAPTRAN-GENFLO allows more accurate prediction of fuel behaviour during transients. Upgrading of the fuel models in reactor dynamics codes has been started, but more work is needed to fully integrate and test new improved models in the production versions of the codes.

Improving the hydraulics models of the reactor dynamics codes remains a challenging task, both scientifically and computationally, and continued work is needed in this field to fulfil the inherent promises of the new models.

Besides now completed tasks, the project has included long term development and validation tasks that will need further work.

The project has continuously contributed to increasing and maintaining nuclear know-how in Finland by educating new experts and transferring information through international organizations and co-operation.

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8.2 Validation of TRAB-3D

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Abstract

TRAB-3D is a reactor dynamics code with three-dimensional neutronics coupled to core and circuit thermal-hydraulics. The code, entirely developed at VTT, can be used in transient and accident analyses of boiling (BWR) and pressurized water (PWR) reactors with rectangular fuel bundle geometry. The validation history of TRAB-3D includes calculation of international benchmark exercises, as well as comparisons with measured data from real plant transients. The most recent validation case is a load rejection test performed at the Olkiluoto 1 nuclear power plant in 1998 in connection with the power uprating project. The fact that there is local power measurement data available from this test makes it a suitable case for three-dimensional core model validation. The agreement between the results of the TRAB-3D calculation and the measurements is very good.

Introduction

Code development in the fields of reactor physics and dynamics, as well as thermal hydraulics has been one of the key areas of reactor safety research in Finland since the middle of the seventies. In dynamics applications the coupling of neutron-physical phenomena with thermal hydraulics of the whole reactor cooling circuit is of vital importance. It has therefore been a built-in property of the Finnish reactor analysis codes from the first applications to the present day 3D models. Major part of the code development work has been carried out by VTT.

TRAB-3D combines the earlier development work into one code, which includes a 3D neutronics core model, together with core and circuit thermal hydraulics models. The development and validation of TRAB-3D has been carried out mostly within the FINNUS programme and its predecessor (RETU). The models of TRAB-3D are based on those of HEXTRAN [1] for the core models, and TRAB [2] for circuit and system models. Both the hexagonal 3D neutronics code HEXTRAN and the 1D dynamics code TRAB have long been used in safety analyses of Finnish and foreign nuclear power plants.

This article gives first a short description of the TRAB-3D code itself, followed by a summary of the validation of the code. As a previously unreported calculation example, some preliminary results of the calculation of the load rejection test performed at the

Olkiluoto 1 nuclear power plant in June 16th, 1998 are given. This test was a part of the power uprating project of the plant.

TRAB-3D Code

The two-group neutron diffusion equations are solved in TRAB-3D by a nodal expansion method in x-y-z geometry within the reactor core [3]. A basic feature of the method is decoupling of the two-group equations into separate equations for two spatial modes and reconstruction of group fluxes from characteristic solutions to these equations. The two solutions are called fundamental or asymptotic mode which have a fairly smooth behavior within a homogenized node, and transient mode which deviates significantly from zero only near material discontinuities.

The nodal equations are solved with a two-level iteration scheme where only one unknown per node, the average of fundamental mode, is determined in inner iterations. The nodal flux shapes are improved in outer iterations by recalculation of the coupling coefficients. Cross sections are computed from polynomial fittings to fuel and coolant temperature, coolant density and soluble boron density.

Thermal-hydraulic calculation of the reactor core is performed in parallel one-dimensional hydraulic channels that are usually coupled with one fuel assembly each. Channel hydraulics is based on conservation equations for steam and water mass, total enthalpy and total momentum, and on a selection of optional correlations describing e.g. non-equilibrium evaporation and condensation, slip, and one and two-phase friction. The phase velocities are related by an algebraic slip ratio or by the drift flux formalism. During hydraulic iterations a one-dimensional heat transfer calculation with several radial mesh points and a specified description of the fuel gas gap is made for an average fuel rod at different axial elevations in each hydraulic channel.

Advanced time integration methods are applied in the dynamic calculation. The numerical technique can vary between the standard fully implicit theta method and the central-difference theta method both in the heat transfer calculation for fuel rods and in the solution of thermal-hydraulic conservation equations for cooling channels.

TRAB-3D includes the BWR circuit model containing one-dimensional descriptions for the main circulation system inside the reactor vessel including the steam dome with related systems, steam lines, recirculation pumps, incoming and outgoing flows and control and protection systems.

In PWRs the circuit thermal hydraulics is calculated with a separate model. The combination code used for square lattice cores with a PWR circuit is TRAB-3D(core model only)-SMABRE. The solution method of the SMABRE [4] circuit model is non-iterative and there is only a loose coupling between it and the TRAB-3D core model. The codes are connected by data changing once during a time-step, no iterations are made with the circuit hydraulics.

TRAB-3D validation summary

Table 6. Validation cases of the TRAB-3D code.

Case	*Code system*	*Reactor type*	*Reference*
OECD LWR Core transient benchmarks	TRAB-3D (core)	PWR, BWR	3, 5, 6
OECD PWR MSLB benchmark	TRAB-3D-SMABRE	PWR	7, 8
OECD BWR TT benchmark	TRAB-3D (core)	BWR	Ongoing work
Olkiluoto pump trip	TRAB-3D	BWR	5
Olkiluoto 1 pressurization transient 1985	TRAB-3D	BWR	9
Olkiluoto 1 instability incident 1987	TRAB-3D	BWR	9
Olkiluoto 1 load rejection test 1998	TRAB-3D	BWR	Ongoing work

The validation of TRAB-3D was started with the calculation of the OECD/NEACRP PWR and BWR core benchmarks [3, 5, 6]. The PWR problem is a control rod ejection transient and the BWR problems are a cold water injection and a core pressurization transient. The second step was to include the BWR circuit models, which was done by recalculating a pump trip, a typical BWR transient, for Olkiluoto reactor and comparing the results with earlier 1D TRAB results [5]. The international benchmark activity and code-to-code comparisons were continued with the calculation of the OECD/NEA PWR main steam line break benchmark, in which TRAB-3D was for the first time coupled to the PWR circuit models of the SMABRE code [7, 8]. Recently VTT has participated also in the calculation of the separate core exercise of the OECD/NEA BWR turbine trip benchmark.

The other line of the validation work has been the comparison of calculated results with plant measurement data. Three such cases, all for the Olkiluoto 1 plant, have been calculated. The first two of them were the pressurization transient in 1985 and the oscillation incident in 1987 [9], the third and most recent being the ongoing calculation of the load rejection test in 1998. The validation work is summarized in Table 6.

Olkiluoto 1 load rejection test 1998

The load rejection test of June 16th 1998 was selected from the tests made in connection with the Olkiluoto plant power uprating, because data on local power measurements was available for this test, making the test a good case for validating 3D core models. The purpose of the test was to ensure that the plant is able to shift from full 2500 MW power operation to a 30 % power level in a case of external load rejection. The plant should be able to feed the in-house load and dump the excess steam to the condensers. This is managed through a partial scram, where one scram rod group is inserted hydraulically in a few seconds and another one slowly electrically, the latter taking 260 seconds. Simultaneously the main circulation pump speed is decreased to the minimum level. In the test the plant functioned as planned.

The test has been calculated with TRAB-3D to 400 s from the beginning, which makes it, by far, the longest BWR transient calculation with VTT's dynamics codes until now. In the calculation the main interest is in the 3D effects in the core. The coolant circuit model was adopted from most recent Olkiluoto calculations as such. Measured values for the feedwater flow and temperature were used as transient boundary conditions.

The steady-state comparisons between the TRAB-3D power distribution and the one reconstructed from measurements show a maximum deviation of 4 % for some individual fuel bundles, while for most bundles the deviation is clearly less, as can be seen in Figure 42. Thus the calculated steady state is close to the real initial state of the reactor.

Figure 43 shows the calculated local neutron flux behaviour against measured data from local power range monitors (LPRM). The calculated value in the figures is an average of the thermal neutron flux of the four bundles surrounding the detector, there are no models for the actual detector response. The data is normalized under the assumption that the initial steady state values are correct. The agreement of the dynamic behaviour is remarkably good.

Average neutron flux behaviour against the data from one of the average power range monitor (APRM) measurement systems is shown in Figure 44. Both the APRM signal and the calculated value are an average of 28 local detector signals at different radial and axial positions in the core.

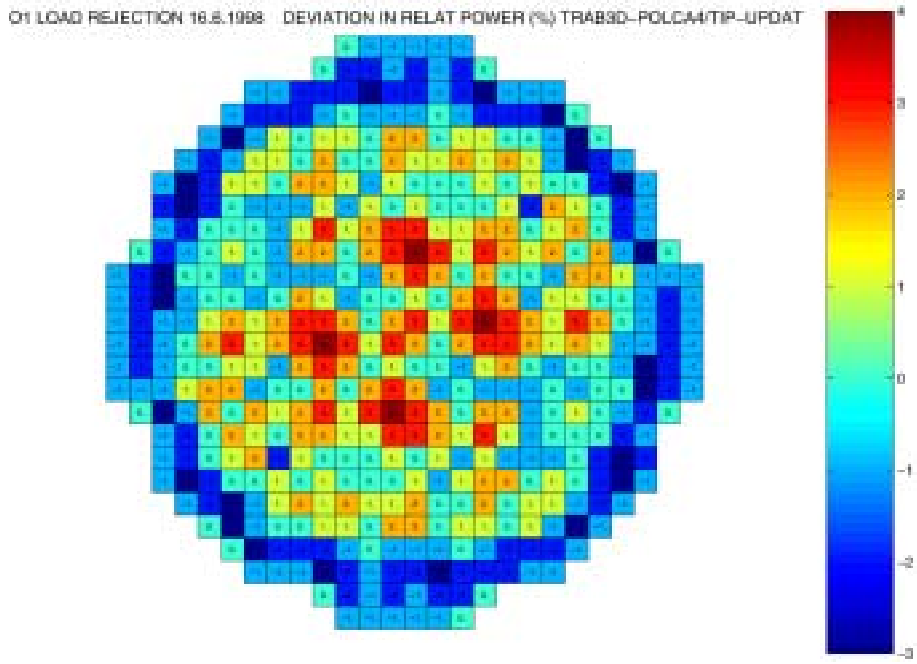

Figure 42. TRAB-3D power distribution deviation (%) from the power distribution reconstructed from measurements.

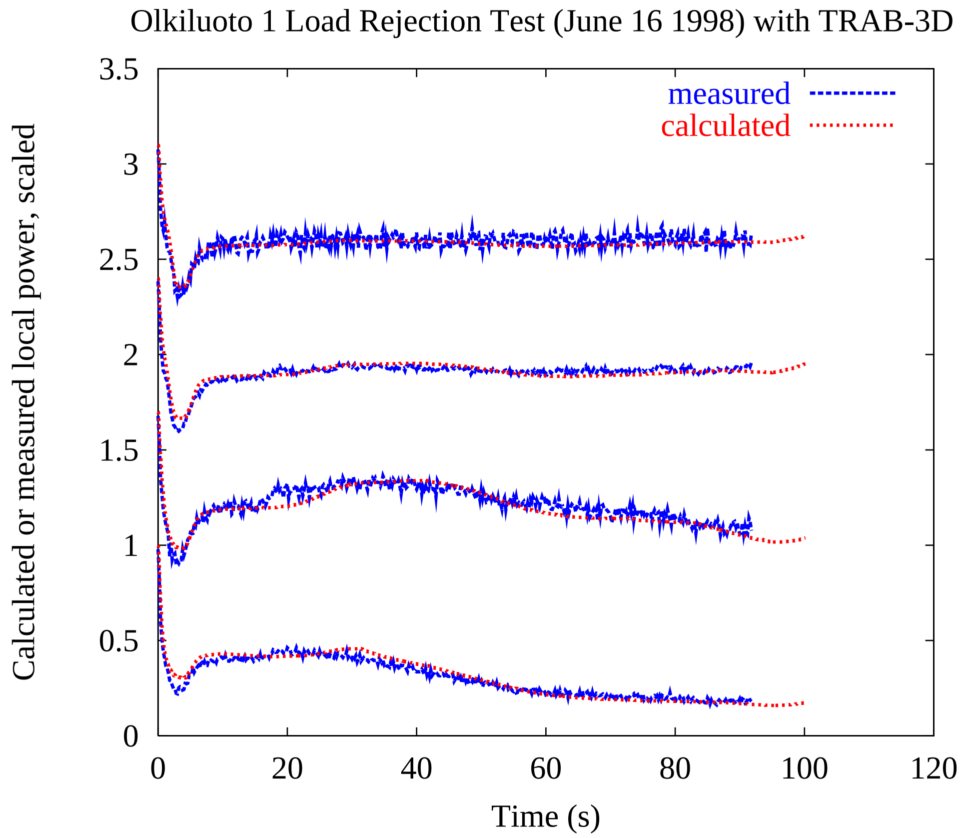

Figure 43. Calculated and measured local power at four axial locations.

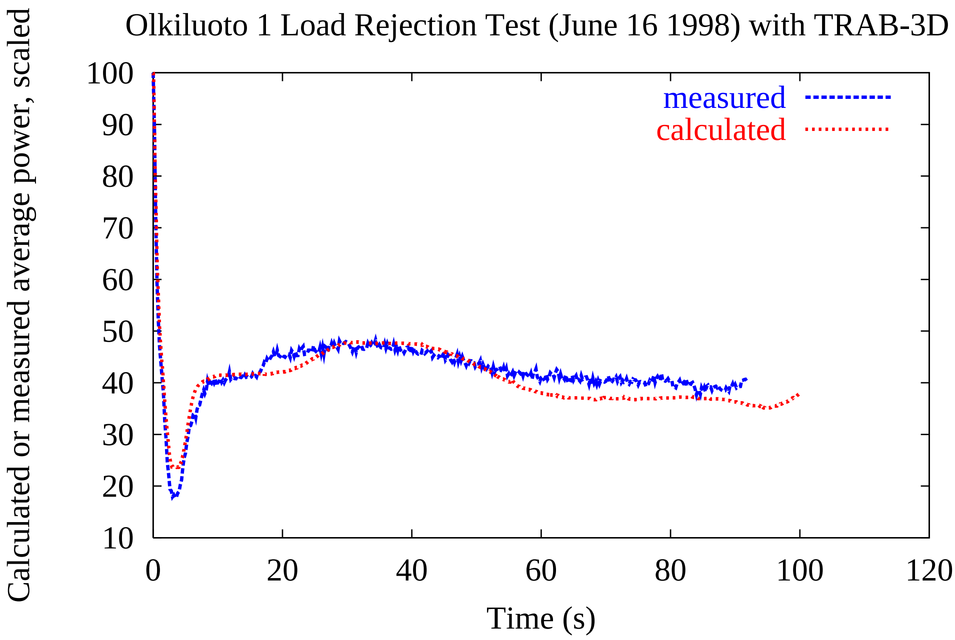

Figure 44. Calculated and measured average power, based on 28 local power measurements at 7 radial locations, each with 4 axial locations.

Conclusions

Preliminary results of the calculation of the load rejection test indicate the TRAB-3D is capable of calculating correctly the dynamic 3D power distribution. In earlier cases, only steady state power distributions have been compared to measurements or results from other computer codes.

From this, together with the good results of the earlier validation cases, it is possible to say that the TRAB-3D code is ready for licencing applications. Until now, the transient analyses for BWRs have been done with one-dimensional models. A new methodology, needed in proceeding to the more realistic three-dimensional analyses, is being studied as a Nordic co-operation effort within the NKS research programme framework.

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8.3 ARES – a new BWR simulator

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Introduction

Coupling the three-dimensional Analytic Function Expansion Nodal (AFEN) nodal model [1, 2] developed within the READY project in 2000–2001 with a four-equation based stationary-state thermalhydraulics module [3] and a new cross section model [4], a basis has been created for a sophisticated BWR simulator code. Motivation for distilling the efforts into a new simulator, named ARES (**A**FEN **Re**actor **S**imulator), can be summarized in three main points:

- Stationary-state analyses required by the safety authorities must be independent from the calculations made by the power utilities. Using a different simulator for some calculations is an effective method for obtaining independent results.

- In order to keep up with the development in the core analysis field, a "test bench" is required for testing and evaluating new ideas and models. In addition, accuracy of the commercial codes and the models incorporated in them can be evaluated by benchmarking them against the new simulator.

- Writing a new program from scratch is potentially a good way to transfer experience from the first generation of Finnish nuclear engineers. It also gives the opportunity to re-evaluate some of the ideas used in the older codes written in times of significantly smaller computer capacity.

Main parts of the code

A boiling water reactor simulator has four main parts:

1. Neutronics module calculating the neutron flux (and power) distribution with given cross section data.

2. Thermalhydraulics module calculating the thermalhydraulic state (water temperatures, void fractions etc.) corresponding to a given power distribution.

3. Cross section module giving the cross section data corresponding to a given thermalhydraulics state using precalculated single-assembly data sets.

4. Burnup module calculating the changes in historic parameters over the burnup cycle.

In development of the ARES code, main effort has been cast in the neutronics module, which uses a rather complicated 38-component analytic base function representation for the intranodal flux shape. In addition to potentially very accurate (within the limitations of diffusion theory) global solution of the neutron flux distribution, it also enables straightforward pin power reconstruction, since the flux values at the node edges are known in addition to the node and boundary averaged values. In order to keep computing times as low as possible, a nodal rebalancing system was developed to cut the computing times from a direct Gauss-Seidel solution by a factor of 20. Still, the amount of data and the complicated nodal couplings tend to make the solution about 10 times slower than on conventional nodal models. However, since the main use of the code will not be in refuel planning, the current speed is acceptable.

The thermal hydraulics module was made on the basis of the TRAWA four-equation model [5], by using second-order discretization of the time-independent conservation equations for water phase mass, steam phase mass, total enthalpy and total momentum. The current method of solution is rather straightforward, and the optimization between speed and stability has been made in favour of the latter.

The cross section module was developed together with the scripts needed to generate the data sets. The approach is a hybrid between table interpolation and polynomial fitting, using polynomial fits with respect to the time-critical momentary quantities (void fraction, moderator and fuel temperature etc.) and second-order table interpolation with respect to the history parameters (exposure, void history etc.) About 2000 state points are needed to generate the cross section data for one material.

The burnup module is not yet finished at the time of writing this presentation. The method of solution has been chosen to be second-order predictor-corrector method using the classical Adams-Bashforth-Moulton scheme.

Validation calculations

The AFEN neutronics model was tested in two benchmark calculations. The first one was the IAEA-3D benchmark [6] representing a 177 assembly pressurized water reactor. Figures 45 and 46 show the thermal flux on two planes of the core. The infinite multiplication factor obtained with the AFEN code was 1.02910, whereas the VENTURE reference calculations gave the extrapolated result 1.02903. The second set of calculations was an analysis of a simplified core of a BWR in cold state. Several calculations were made with different control rod patterns. Reference results were obtained with the SIMULATE-3 nodal code that uses a separable fourth-order polynomial model for the intra-nodal flux.

Some results of these calculations are shown in Table 7. The first two cases are ”calibration calculations” with a homogeneous core. Cases 3, 4 and 5 have different control rod patterns. Cases 3 and 4 were repeated with assembly discontinuity factors set to unity – these are referred to as cases 3b and 4b. In addition, SIMULATE-3 calculations of case 4 were repeated with some improved models regarding e.g. the calculation of assembly discontinuity factors. Table 7 shows the multiplication factor and nodal power peaking in each case. S refers to SIMULATE-3, A to AFEN and S+ to SIMULATE-3 with the improved models switched on.

Table 7. Results of cold-state BWR test calculations.

	k-eff, S	k-eff, A	k-eff, S+	peak, S	peak, A
Case 1	1.11072	1.11081		3.339	3.343
Case 2	0.94543	0.94548		3.268	3.295
Case 3	0.96863	0.97073		40.66	43.43
Case 3b	0.96302	0.96365		31.76	32.34
Case 4	0.99101	0.99238	0.99267	15.43	16.02
Case 4b	0.98537	0.98557		13.26	13.40
Case 5	0.99132	0.99267		14.55	15.13

The coupled neutronics, thermal hydraulics and cross section modules were tested by repeating the BWR calculations, this time at full power state. The void distribution at the core cross section in one such case is shown in Figure 47. These calculations have not yet been repeated with another simulator, and the main focus has been in checking the overall sensibility of the results. A more thorough set of test calculations is planned after the burnup module has been completed. This is scheduled to take place in the last months of this year.

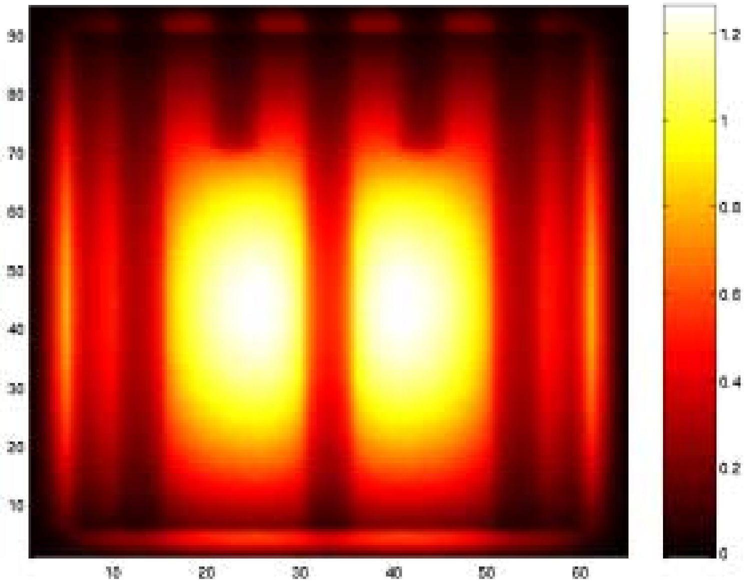

Figure 45. Thermal flux in a cross section of the IAEA-3D benchmark problem.

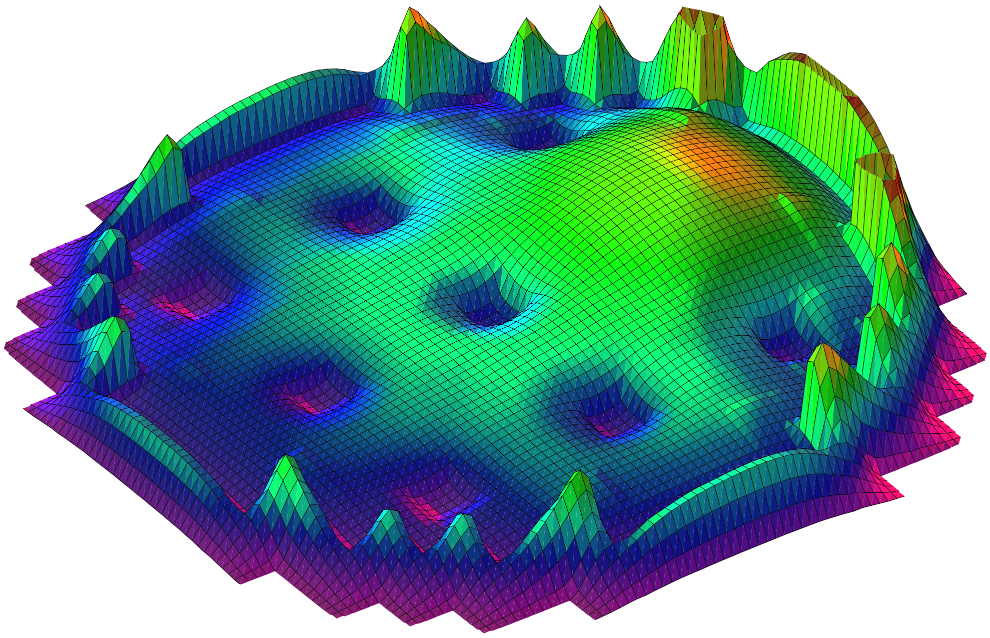

Figure 46. Thermal flux on the midplane of the IAEA-3D benchmark problem.

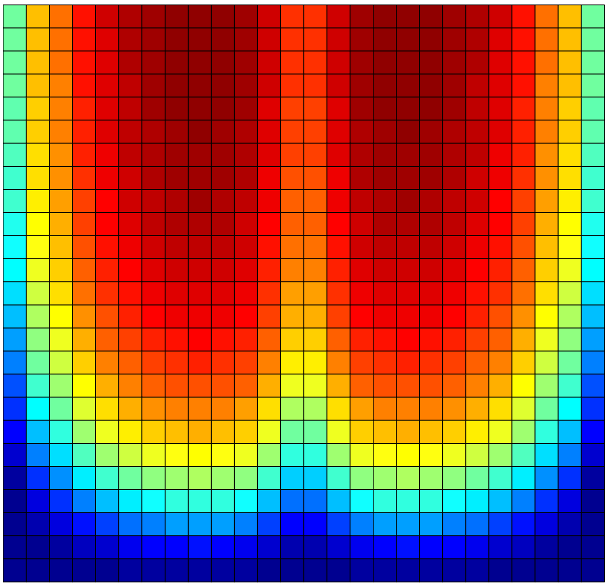

Figure 47. Void distribution in a cross section of a BWR test problem.

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9. Thermal-hydraulic experiments and code validation (TOKE)

9.1 TOKE summary report

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Abstract

The thermal-hydraulic experiments and code validation (TOKE) project addressed both the experimental and computational aspects of nuclear safety studies. Integral VVER related experiments dealing with a steam generator collector header rupture incident and with non-condensable gas behaviour in the primary circuit were carried out in the PArallel Channel TEst Loop (PACTEL). Local loading effects due to water flow and thermal stratification in a T-joint of a hot horizontal pipe and a cold vertical tube were investigated in a purpose-built test loop in co-operation with the structural integrity (STIN) project. The behaviour of non-condensable gas during the first seconds of a conceivable large break loss-of-coolant accident (LBLOCA) blowdown to a boiling water reactor (BWR) condensation pool was also studied in the separate effect tests related subproject. For this purpose, a test rig with a scaled down water pool, blowdown pipes, an emergency core cooling system (ECCS) strainer and a pump was designed and constructed in Lappeenranta University of Technology (LTKK). Thermal-hydraulic and computational fluid dynamics (CFD) calculations with the codes APROS and Fluent, respectively, supported the planning and analysis of both the integral and separate effect tests.

Introduction

The significance of fundamental understanding of thermal hydraulics for successful and safe plant operation in all conceivable conditions is clear. For example, a great number of thermal hydraulic processes have to be taken into consideration when prevention strategies and mitigation measures are planned for different accident scenarios. Experimental investigation is one of the means to address the research needs in the field of thermal hydraulics. However, plant scale experiments are often impossible or too expensive. Scaled down test facilities give an opportunity for system level as well as for separate effect thermal hydraulic studies. Another approach to thermal hydraulics is to develop simulation codes. The safety of nuclear reactors from thermal hydraulics point of view can be computationally assessed reliably, if the system analysis codes used are adequately validated for application to a large variety of accident scenarios. Experimental work on test facilities contributes significantly both to the basic understanding of phenomena occurring during abnormal situations and accidents in nuclear reactors as well as to the assessment of safety analysis tools.

The target of the TOKE project was to produce thermal-hydraulic data for developing more accurate safety analysis tools and to enhance understanding of phenomena occurring during abnormal situations in nuclear reactors. The project consisted of three parts: 1) VVER related integral tests with the PACTEL test loop [1], 2) thermal-hydraulic and fluid dynamics computer code calculations supporting the tests and 3) separate effect tests with purpose-built facilities.

Main objectives

In the area of VVER related tests with the PACTEL test facility, the main goal was to examine a steam generator collector header rupture incident. At the Loviisa nuclear power plant (NPP), the construction of the collectors was changed. In the case of a collector header rupture, the leak flow area from the primary to the secondary side was reduced significantly. The effect of the construction change was studied with two experiments in PACTEL. Another major VVER related task was the test series on non-condensable gases. The effects of noncondensables on the heat transfer of a horizontal steam generator at full and reduced primary side coolant inventory were studied.

In the thermal-hydraulic and CFD calculations subproject, the main efforts were to validate the APROS code against OECD/NEA Main Steam Line Break benchmark (TMI-I) and against PACTEL results, to support separate effect test planning with the Fluent code calculations of a gas bubble behaviour in a condensation pool and to produce in-kind contribution for the CAMP agreement.

The separate effect tests related subproject focused on constructing a test loop and carrying out experiments to study local loading effects due to water flow and thermal stratification in a T-joint of a hot large diameter horizontal pipe and a cold small diameter vertical tube. Using the measured process and boundary conditions data the loading transients were calculated with a CFD code in the STIN project. Finally, the produced loading data was used for structural analysis. Another major objective of this subproject was to study the behaviour of non-condensable gas in a BWR condensation pool during a LBLOCA blowdown. For this purpose, a large-scale test facility consisting of a water pool, blowdown pipes, an ECCS strainer and a pump was built. Experiments on gas bubble formation and behaviour in the pool right after the initiation of blowdown and on the pump performance as gas flows through it were carried out. The third separate effect test facility was a small-scale, plexiglass experimental set-up modeling the top of a VVER reactor upper plenum. Thermal stratification during natural circulation cooldown was investigated with a series of tests and CFD calculations.

Thermal stratification in piping system dead legs

Thermal stratification of hot and cold water can take place in different systems and components in power plants. Moving or cycling stratification layer may cause thermal loads to structures and induce leakages in pipes. The reason for cyclic behaviour may be for example a leaking valve. A test loop for carrying out experiments to study thermal stratification and loads in a T-joint of a large diameter (ID=243 mm) horizontal main pipe and a small diameter (ID=50 mm) vertical dead leg was designed and constructed in LTKK [2]. The experimental set-up was based on the geometry of the connection line between hot and cold legs at the Loviisa NPP, where cracks near the T-joint caused a leak from the primary circuit [3]. In the test loop, turbulent, hot (250 °C) flow in the horizontal pipe and cold (25–50 °C) stagnant or low velocity fluid in the vertical dead leg were used to produce cyclic temperature behaviour near the T-joint connection. An illustration of the vertical dead leg below the main pipe is shown in Figure 48.

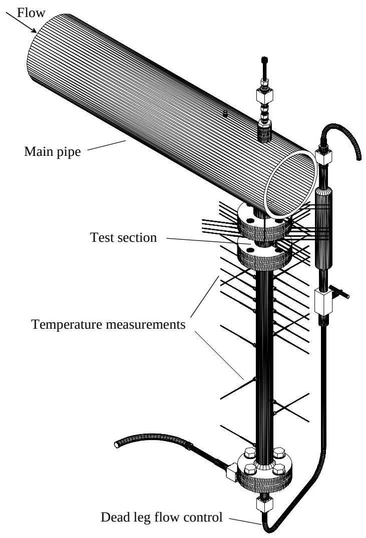

Figure 48. Experimental set-up in thermal stratification tests.

Turbulence from the main pipe penetrates into the dead leg where a vortex is formed. Temperature is highest on the downstream side, where hot water from the main pipe flows into the dead leg and lowest on the upstream side, where water returns towards the main pipe. The inner wall temperature distribution of the dead leg, when the flow velocity in the main pipe is 0.56 m/s ($Re=1.03x10^6$) and the fluid in the dead leg lower end is stagnant, is shown in Figure 49. The length of the vortex is estimated to be y/D_{dl}=2.0–2.5, where y is the length and D_{dl} the inner diameter of the dead leg. At y/D_{dl}=2.7, the temperature is practically constant across the dead leg indicating that the vortex has died away. The steepest change in wall temperature occurs close to the lower end of the vortex in the transient and laminar region, where the heat transport decreases drastically. Further away from the T-joint the stratification decreases and becomes negligible at about 5 diameters. The situation does not change much, if the velocity in the main pipe is smaller (0.27 m/s, Re=4.97×10^5). The maximum temperature difference between the downstream and upstream side of the dead leg is about 10 ºC, the same as in the high velocity case. Only the length of the vortex is now shorter.

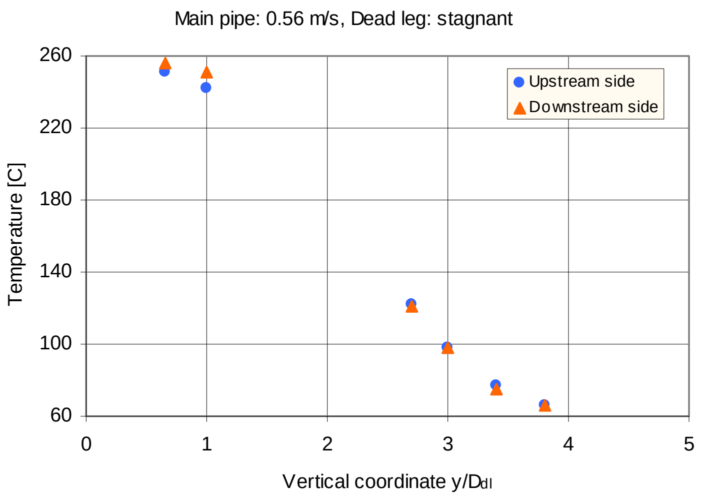

Figure 49. Inner wall temperatures of the dead leg in thermal stratification tests.

With a small outflow (0.9 mm/s, about 6.2 litres per hour) from the lower end of the dead leg the results are qualitatively different from the stagnant flow situation. Fluid at a temperature between 100–200 ºC occupies the upper half of the dead leg. As the hot water flows downwards through the uninsulated dead leg, it is cooled by heat convection and radiation to the surrounding room. As a result, the temperature gradient becomes smoother and the stratification front stretches.

Selected thermal stratification test results were used in the STIN project for verifying the capability of CFD calculations to predict the stratification [4]. The results were utilized further in the development and assessment of a tool for transferring data from CFD calculations to structural analysis [5]. Vertical and circumferential strains on the test section outer surface were measured during the experiments to produce comparision data for structural analysis.

Studies on non-condensable gas blowdown to a BWR suppression pool

During the first seconds of a conceivable LBLOCA in a BWR a large amount of non-condensable gas (nitrogen) will be blown from the upper drywell to the condensation pool through the blowdown pipes. Gas discharging to the pool could push its way to the ECC systems and diminish their performance or even break the ECC pumps. A test rig with a scaled down water pool and an ECCS strainer modeling the condensation pool of the Olkiluoto NPP was designed and constructed in the TOKE project. The behaviour of non-condensable gas (air) bubbles during blowdown and the effect of gas on the performance of an ECCS pump were studied. CFD calculations were made to support the design of the test rig and the planning of the experiments.

The first large air bubbles forming at the blowdown pipe outlet touched the ECCS strainer. Later, air rose to the surface around or very close to the blowdown pipe leaving the air volume fraction small elsewhere in the pool. The rising air plume lifted water up to the surface, which was splashing strongly. A circulation pattern was created. Water ran back down close to the wall of the pool and carried down smaller air bubbles. When two blowdown pipes were used simultaneously, a lot of air bubbles were detected inside the strainer during the first 30 seconds.

In the ECCS pump tests; the objective was to measure the gas volume fraction in the flow that drops the performance of the pump significantly. The pump was running at nominal speed and pumping water with four different flow rates when air was injected directly into the intake pipe. The air volume fraction was increased step-by-step until the flow and head of the pump collapsed. The results show that the threshold value for air volume fraction is 3–7 % depending on the water flow rate (Figure 50).

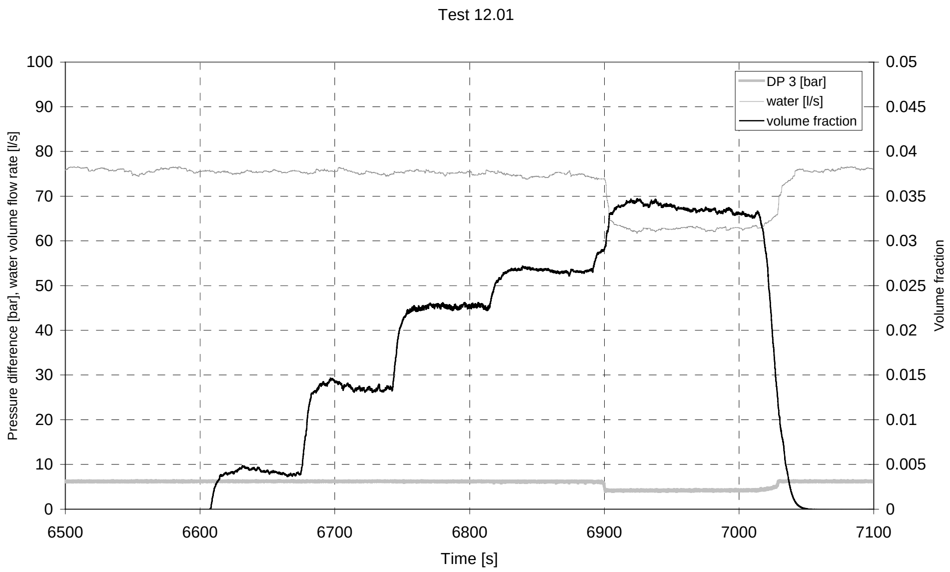

Figure 50. ECCS pump's response to the presence of non-condensable gas in the flow.

VVER system behaviour during a steam generator collector header rupture incident

The work on collector header rupture incident focused on assessing the effect on the system behaviour of the construction change made to the steam generator primary collectors at the Loviisa NPP. As a result of the change, the leak flow area from the primary to the secondary side was reduced significantly in the hypothetical case of a collector header rupture. Two experiments were carried out in the PACTEL facility [6]. The experiment procedure followed the regulations for operator actions during a state of an emergency in force at that time at the Loviisa NPP. With the larger break size, the occurring phenomena were faster and the safety valve of the broken steam generator cycled few times releasing secondary side inventory to the atmosphere (Figure 51). The steam generator safety valve didn't open if the flow restrictor (small break) was used.

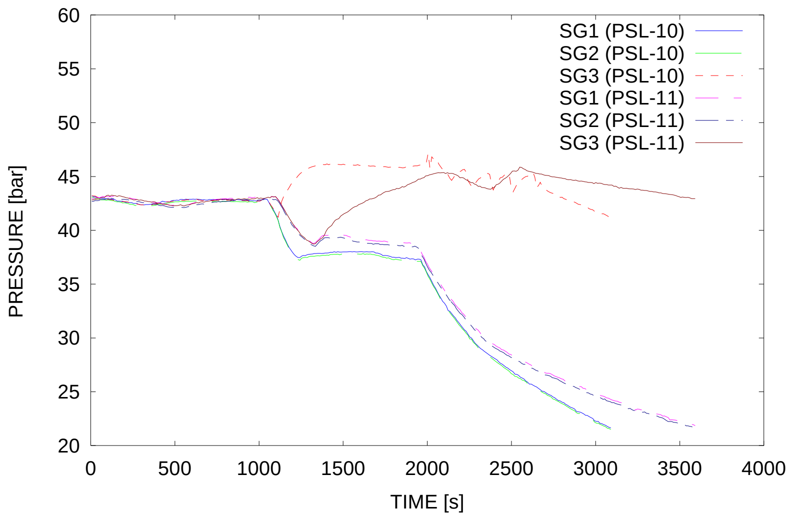

Figure 51. Secondary pressures with (PSL-11) and without (PSL-10) the flow restrictor in the steam generator collector header rupture experiments.

PACTEL experiments and APROS simulations on non-condensable gas behaviour in the VVER primary circuit

Noncondensables can have an effect on system behaviour, for example, during LOCAs if accumulator gas (nitrogen) is released into the primary circuit piping. The effect of non-condensable gas on system thermal-hydraulics and on heat transfer in a horizontal steam generator at full and reduced primary coolant inventory was studied in the PACTEL facility [7]. Air and helium acted as non-condensable gases that were injected into the vertical section of the hot leg below the steam generator.

With full initial coolant inventory the system response to the introduction of non-condensable gas was as expected. Gas accumulated to the uppermost tubes of the steam generator and diminished its heat transfer capacity. This led to an increase of primary pressure, which in return compressed the gas volume at the top of the tube bundle. As a result, a new system steady state with improved heat transfer was found.

With reduced inventory i.e. in two-phase natural circulation or boiler condenser modes, the system response depended on the gas in question and was not so coherent. Air accumulated mostly to the bottom tube rows of the steam generator while vapour, being lighter than air, continued to flow through the top part of the bundle and condense there (Figure 52). The measured loop behaviour with helium differed less than expected from the case, where air was used. Regardless of being lighter than vapour, helium did not accumulate clearly to the uppermost tubes. Instead, vapour and helium mixed with each other and the heat transfer deteriorated rather uniformly across the tube bundle.

The helium test was simulated with the APROS 5.02 system code [8]. The material properties of air (heat capacity and conductivity, mole weight) were changed to correspond with the helium properties. The Vierow-Schrock correction factor was changed to use gas mole fraction instead of mass fraction. The Shah correlation for condensation heat transfer was used. The PACTEL simulation model was based on reference [9].

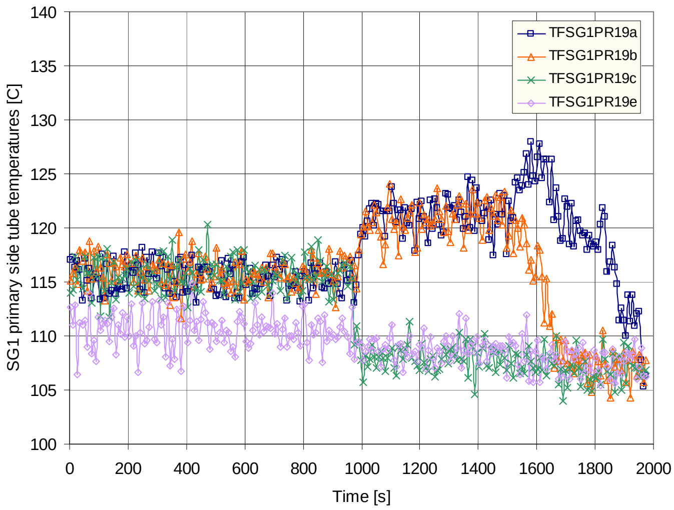

Figure 52. Heat transfer deterioration at the elevation of 0.25 m from the steam generator bottom due to the injections (at 960 s and 1500 s) of non-condensable gas (air) as indicated by temperatures along a single tube.

The numerical simulation of the helium-water mixture turned out to be difficult. Especially, problems in calculating condensing heat transfer in horizontal pipes were encountered. In the steady state phase, the calculated heat transfer is too small. After the helium injection, it is too large. As a result, the measured and calculated temperatures close to the inlet of the uppermost tubes differ from each other. In the bottom tubes, the measured and calculated temperature distribution is closer to each other (Figure 53).

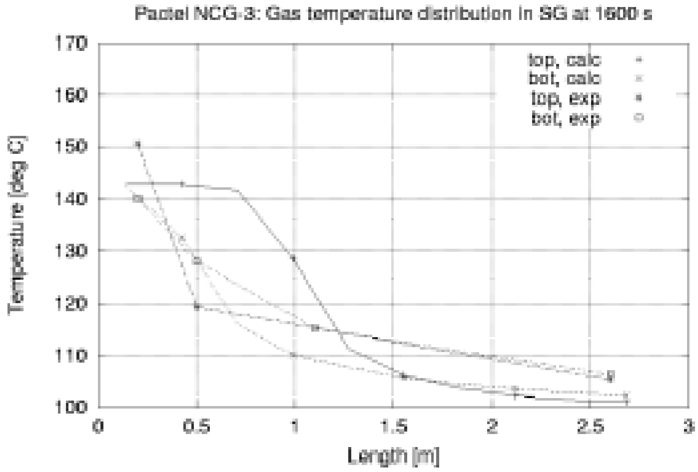

Figure 53. Calculated and measured gas temperature distributions inside the top and bottom steam generator tubes at 1600 s in the helium gas experiment with PACTEL.

According to the simulation, 75 % of helium remained in the steam generator after the first injection. The rest 25 % was found in the cold leg. After the second injection, some helium appeared also at the top of the downcomer. The calculated helium distribution in the system differed from that of the experiment and this may have contributed to the differences in the steam generator heat transfer.

TMI-I main steam line break benchmark

In the simulation of the OECD/NEA Main Steam Line Break benchmark (TMI-I), exercise 1 was calculated with the APROS 4.06 point kinetics model and the results were compared with the SMABRE calculations by VTT in the READY project and with the TRAC calculations by the benchmark organizers (PSU) [10]. The SMABRE results were very close to the TRAC results. The main difference between the APROS and SMABRE results is in the break flow rate. In the SMABRE calculation, more liquid was carried with the steam flow to the break and the steam generator dried out earlier. In the APROS calculation, the steam flow to the break lasted longer and about 7000 kg more steam was released through the break. The higher steam release in the APROS calculation cooled down the plant longer. This resulted in lower core temperatures and in a power increase that lasted longer.

Applications

Valuable expertise in designing and constructing test facilities that are applicable to safety studies of both Olkiluoto and Loviisa NPPs was gained through the TOKE project. The PACTEL facility, the thermal stratification test loop and the condensation pool test rig were extensively used for reducing uncertainties in the understanding of important thermal hydraulic phenomena. PACTEL integral results can be utilized, for example, in verifying the effects of modifications made to the VVER type reactors, such as the construction change of the steam generator primary collectors at the Loviisa plant. The scaled down condensation pool test rig gives an opportunity to study BWR related transients of the current plant generation as well as passive safety systems of the next plant generation. Studying the behaviour of a single component succeeds also, as evidenced by the experiments carried out, to determine the threshold value for the safe air volume fraction in the flow through an ECCS pump of the type used in BWRs.

In parallel with experimental thermal hydraulics, the development of simulation codes contributes to the safety of nuclear reactors. Thermal hydraulic and CFD codes are already applicable to a large variety of accident scenarios. However, a new application of the APROS system code to the calculation of thermal hydraulic phenomena was tested in the TOKE project. Previously, cases where helium is the non-condensable component had not been simulated with APROS. The necessary material properties were changed and a new correlation for non-condensable gas was implemented. Also, the ability of CFD codes to simulate the behaviour of non-condensables in a large water pool was confirmed with Fluent 5 calculations of the condensation pool experiments.

A tool for transferring loading data produced by CFD calculations to structural analysis was developed and assessed in the STIN project with the help of the thermal stratification test results. The research group gained also useful knowledge of performing structure stress measurements using temperature compensated strain gauges.

Conclusions

The VVER related tests carried out in the TOKE project aimed at studying integral system behaviour during transients and accidents. The effect of the construction change made to the steam generator primary collectors at the Loviisa NPP was verified with two PACTEL experiments. System thermal-hydraulics and heat transfer in a horizontal steam generator at full and reduced primary coolant inventory and in the presence of non-condensable gas was studied with a series of single loop experiments in the PACTEL facility. The system response depended on the gas in question but was not entirely what was expected. A need for more detailed experiments was manifested.

Moving or cycling stratification layer may cause thermal loads to structures and induce leakages in pipes. Process and boundary condition data with the help of a purpose-built test loop was produced for CFD calculations of loads caused by water flow and thermal stratification. The calculated thermal loads were then used as input to structural analysis. Valuable co-operation between the thermal hydraulics, numerical simulation and structural analyses branches of the FINNUS research programme was created.

A test rig modeling the condensation pool of the Olkiluoto NPP was designed and constructed to study the behaviour of non-condensable gas during the first seconds of a LBLOCA blowdown. The effect of gas on the performance of an ECCS pump was also examined. In the tests, only a small amount of non-condensable gas was observed inside the ECCS strainer or in the ECCS pump flow, when one blowdown pipe was used. With a direct injection of gas into the pump intake pipe the critical gas volume fraction for the safe performance of the pump was found to be 3–7 % depending on the water flow rate.

Thermal-hydraulic and CFD calculations were effectively used in the planning and analysis of different tests in the TOKE project. The simulations gave an opportunity for the researchers to maintain comprehensive computational expertise in a useful way.

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9.2 Condensation pool experiments with non-condensable gas and Fluent 5 simulations

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Abstract

The formation, size and distribution of non-condensable gas bubbles in the condensation pool of the Olkiluoto nuclear power plant (NPP) in a conceivable loss-of-coolant accident (LOCA) was studied experimentally with a scaled down condensation pool test rig. Particularly, it was important to find out if any air bubbles flowed inside the emergency core cooling system (ECCS) strainer close to the pool wall and bottom. The effect of non-condensable gas on the performance of an ECCS pump was also examined. Computational fluid dynamics (CFD) calculations with the Fluent 5 code were made to support the design of the test rig and the planning of the experiments. Compressed air was blown to the test pool through blowdown pipes or, alternatively, air was injected directly into the intake pipe of the ECCS pump. The first large air bubbles forming at the blowdown pipe outlet touched the ECCS strainer. When two blowdown pipes were used simultaneously, a lot of air bubbles were detected inside the strainer during the first 30 seconds. A 3–7 % volume fraction of air injected directly into the pump intake pipe was enough to make the pump head and flow collapse.

Introduction

During a LOCA a large amount of non-condensable (nitrogen) and condensable (steam) gas will be blown from the upper drywell to the condensation pool through the blowdown pipes at the Olkiluoto NPP. There might be a risk that the gas discharging to the condensation pool could push its way to the ECC systems and diminish their performance or even break the ECCS pumps. A test rig with a scaled down water pool and a pump was designed and constructed in Lappeenranta University of Technology (LTKK) for carrying out a series of tests under this topic. The objective was to study the behavior of non-condensable gas bubbles in the condensation pool and the effect of gas on the performance of an ECCS pump. Previously this problem has been analyzed at VTT Processes by hand calculations and numerically with the Fluent CFD code [1].

Condensation pool test rig

The main components of the test rig were a condensation pool (inner diameter 2.4 m, height 5.0 m), two pressure tanks, two blowdown pipes, an ECCS strainer, an ECCS pump and a compressed air injection system connected to the pump intake pipe. The blowdown pipes were positioned asymmetrically in relation to the centre of the pool bottom. Instrumentation included temperature, pressure and flow rate measurements. The events in the pool were recorded with several video cameras through windows on the pool walls and from above with an underwater camera. The compressed airflow from the pressure tanks to the condensation pool through the blowdown pipes simulated the nitrogen gas flow of a LOCA. Figure 54 shows a 3D-figure of the test rig.

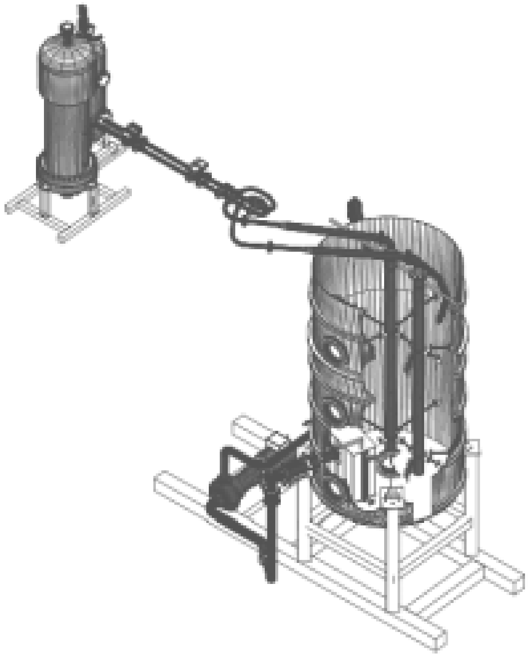

Figure 54. Condensation pool test rig.

Experiments

Experiments were carried out in four different stages:

1. Experiments with blowdown pipes only
2. Experiments with blowdown pipes and an ECCS strainer
3. Experiments with blowdown pipes, an ECCS strainer and a pump (integral tests)
4. Experiments with an ECCS pump and non-condensable gas.

In the first stage experiments, compressed air was blown from the pressure tanks to the pool filled with water through the blowdown pipes. Two different blowdown pipe diameters were used, DN150 and DN200. The idea was to find out the shape and maximum size of the first air bubbles with each blowdown pipe diameter. The largest bubble diameter (1.6 m) was measured with the DN150 pipe. Figure 55 shows how the gas behavior in the pool developed after the initial bubbles were blown. The rising air lifted a large amount of water up to the surface, which was splashing strongly. Water ran back down close to the wall of the pool. This backflow carried down a lot of smaller air bubbles. After 20 seconds, the pool water cleared and the new bubbles forming at the blowdown pipe outlet started to rise up nicely around the pipe.

In the second stage, the ECCS strainer was installed inside the pool. Compressed air was blown to the pool through DN150 pipes. The purpose of the experiments was to study the distribution of air in the pool and the effect of the strainer on bubble behavior. Particularly, it was important to find out if any bubbles flowed inside the strainer. The upper part of the strainer was inside the first large air bubble forming at the blowdown pipe outlet for a while. The next few air bubbles touched the strainer too. When two blowdown pipes were used simultaneously, a lot of air bubbles were detected inside the strainer during the first 30 seconds.

In the third stage, the ECCS pump and the connection piping from the strainer to the pump were added to the test rig. The objective of the integral tests was to estimate the void fraction in the pump intake pipe during the blowdown and also to find out if the pump works properly under these conditions. The pump was sucking water from the pool through the strainer with a constant flow rate (5.5 and 11 l/s) when air was blown to the pool through the DN150 pipes. At the flow rate of 5.5 l/s, air bubbles were not detected inside the transparent section of the intake pipe but at the flow rate of 11 l/s, air bubbles were visible approximately for 20 seconds. A rough estimation of the maximum air volume fraction was made from video frames. The result was 5 %. In the integral tests, the small amount of air seeping through the strainer into the pump intake pipe had no effect on the performance of the pump.

In the fourth stage, the performance of the ECCS pump was examined in more detail. The pump, that was used, simulated the containment spray system 322 pump in Olkiluoto plant conditions (rated flow 75 kg/s at the nominal speed of 2970 rpm). The idea was to find out how much air can be injected into the pump intake pipe before the pump stops working properly. In the experiments, the pump was running at the nominal speed of 2970 rpm with four different water flow rates (12.5, 25, 57 and 75 l/s) when air was blown directly into the intake pipe. The flow rate of air was increased step-by-step until the water flow through the pump and the pump head collapsed. The results show that the threshold value for air volume fraction is 3–7 % depending on the water flow rate. After the air injection was switched off, the head and flow normalized quickly back to the original values with one exception. At the smallest flow rate (12.5 l/s), it took about 30 seconds for the pump head to normalize.

The jet hits the bottom of the pool.

The first bubble expands and rises upwards.

Small air bubbles circulate back to the pool bottom from the surface.

After 20 seconds, new bubbles forming at the pipe outlet rise up nicely around the pipe.

Figure 55. Airflow penetrating to the test pool filled with water.

CFD simulations of the condensation pool

During the design phase of the condensation pool test rig, CFD calculations were made to support the design and the planning of the experiments [2]. A commercial CFD code Fluent 5 was used to simulate blowdown of air into a large pool of water. The purpose of the CFD analyses was to study different geometry arrangements (e.g. the submergence of the blowdown pipe, the pipe diameter, the number of pipes) and to evaluate the phenomena during blowdown (e.g. level swell in the pool, the size of the initial bubbles and the distribution of air in the pool).

Initial analyses were made with an axially symmetrical 2D model of the pool. The 2D calculations were relatively fast, allowing a large number of cases to be studied. The 2D calculations were used to determine the parameters for the final 3D simulations. Two 3D simulations were made, one with a single blowdown pipe and another with two blowdown pipes. The ECCS strainer and pump were not considered in the simulations.

The Volume of Fluid (VOF) model of Fluent 5 was used to model multiphase flow. VOF describes two or more immiscible phases. In each computational cell, volume fractions of the phases are solved, and the material properties in the cell are determined based on the void fractions. Only one momentum equation is solved for the phases. In the momentum equation, the effect of the volume fractions is taken into account by the volume fraction dependent density and viscosity.

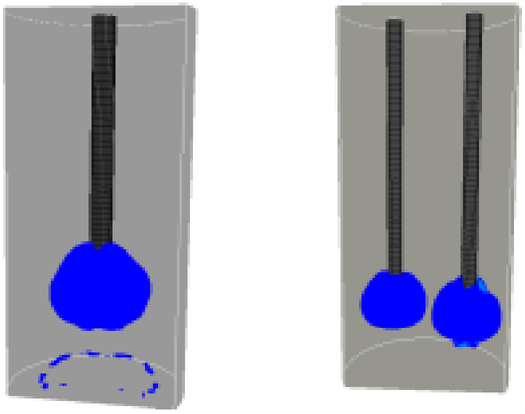

Figure 56. Initial bubbles in 3D simulations.

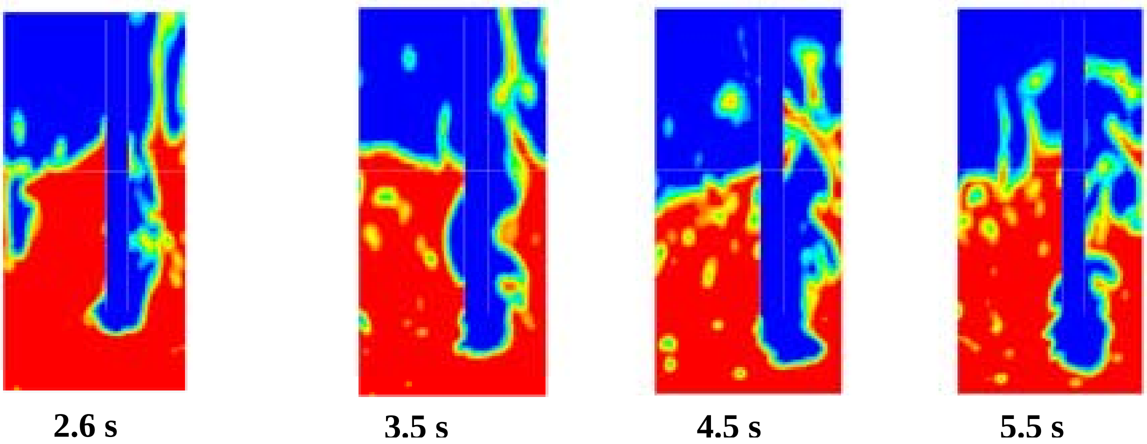

Figure 57. Air distribution at the cross-section going through the pipe centre axis in 3D simulation.

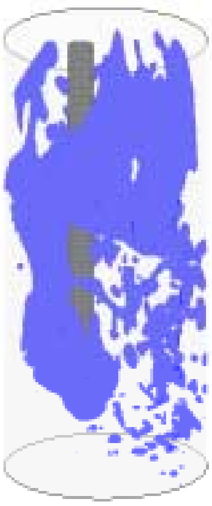

Figure 58. Air distribution in the pool later in the simulation.

The large initial bubbles in the 3D simulations are shown in Figure 56. In the case of a single pipe, a ring of air bubbles can clearly be seen at the bottom of the pool. The ring is formed when the jet from the pipe hits the bottom of the pool. Air rose to the surface of the pool around the pipes, while the air volume fraction elsewhere in the pool was smaller. The surface of the pool was splashing strongly (Figure 57). In the pool, a circulation was created. The rising air plume was entraining water up to the surface around the pipe, while at the periphery of the pool, water was flowing down to replace the water air was carrying up. Smaller air bubbles were transferred with the downward flowing water down to the bottom of the pool (Figure 58).

Conclusions

In this paper, the condensation pool experiments carried out with the test rig designed and constructed in LTKK have been described. The CFD code Fluent 5 was used to support the design of the test rig and the planning of the experiments.

Large air bubbles were formed in the condensation pool when compressed air was blown to the pool through the blowdown pipes. The upper part of the ECCS strainer was inside the first bubble for a while. Also, the backflow of water near the pool walls carried a lot of air bubbles from the surface to the level of the ECCS strainer. The amount of air bubbles that drifted inside the pump intake pipe was, however, so small that the pump head and flow didn't decline during the blows. In the pump tests, it was possible to inject a 3–7 % air volume fraction directly into the pump intake pipe before the head and flow collapsed. At the 12.5 l/s water flow rate, it took about 30 seconds before the head and flow reached the original values again after the air injection was switched off. With other flow rates (25–75 l/s), the head and flow normalized in a few seconds after the air injection was switched off.

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9.3 Experiments on thermal stratification in a piping system dead leg, CFD simulations and structural analyses (TOKE & STIN)

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Abstract

Thermal stratification of hot and cold water in piping system dead legs has been investigated. The work has been done in co-operation between the thermal hydraulic experiments and code validation (TOKE) and the structural integrity (STIN) projects of the FINNUS research programme. The heat transport in the stratification region was studied with temperature measurements, and the response of the structures to the heat loads was investigated with strain gauges. Heat loads on the inner walls of the pipes were calculated with computational fluid dynamics (CFD) and transferred to structural analysis for the determination of stresses and strains. The calculations were validated by comparing the results with the experiments.

Introduction

Thermal stratification of hot and cold water can take place in different systems and components in power plants. Moving or cycling stratification layer may cause thermal loads to structures and induce leakage in pipes. The reason for cyclic behaviour may be for example a leaking valve, vibration of a pipeline or turbulence of the main pipe interacting with a thermally stratified layer in a branch pipe.

A test loop for carrying out experiments to study local loading effects due to water flow and thermal stratification in a T-joint of a hot horizontal pipe and a cold vertical dead leg was designed and constructed in Lappeenranta University of Technology (LTKK) [1]. The experimental set-up was based on the geometry of the connection line between hot and cold legs at the Loviisa Nuclear power plant, where cracks near the dead leg caused a leak from the primary circuit [2].

The test loop consisted of a large hot (250 °C) horizontal pipe with turbulent flow and a vertical cold (25…50 °C) dead leg with stagnant or low velocity flow. The test results were used for verifying the capability of the CFD calculations to predict the stratification. In addition, the stresses and strains calculated with the structural analysis by using the loads of the CFD calculation were compared to the measurements [3].

Experiments on thermal stratification

The test loop of the thermal stratification experiments consisted of an uninsulated vertical dead leg (D_{in} = 49.2 mm) connected to the bottom side of a horizontal main pipe (D_{in} = 242.9 mm) at a distance of 16 diameters from the pipe inlet. The dimensionless length L/D of the dead leg was 24.9. A sketch of the test loop is shown in Figure 59.

Densely positioned thermocouples along the vertical test section were used to investigate the formation and movement of the stratification front. Verification data for structural analysis was gathered with temperature compensated biaxial 2-element type strain gauges mounted on the outer wall surface of the dead leg.

Two different flow velocities (0.27 and 0.56 m/s) in the main pipe were used. In both cases, the flow was fully turbulent with Reynolds numbers 4.97×10^5 and 1.03×10^6, respectively. The flow in the dead leg was either stagnant or it had a small velocity (0.9 mm/s) downwards and a Reynolds number smaller than a few hundred.

Turbulence from the main pipe penetrates into the dead leg where a vortex is formed. Temperature is highest on the downstream side, where hot water from the main pipe flows into the dead leg and lowest on the upstream side, where water returns towards the main pipe. Figure 60 shows the inner wall temperature distribution of the dead leg,

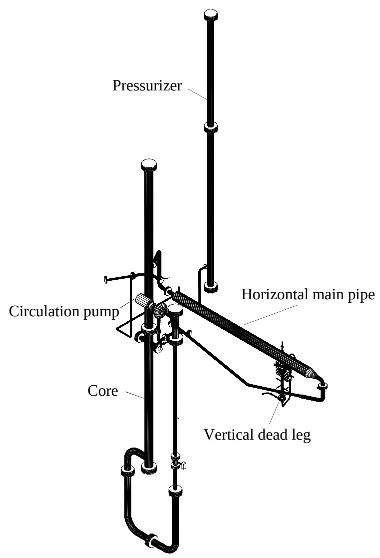

Figure 59. Test loop in thermal stratification tests.

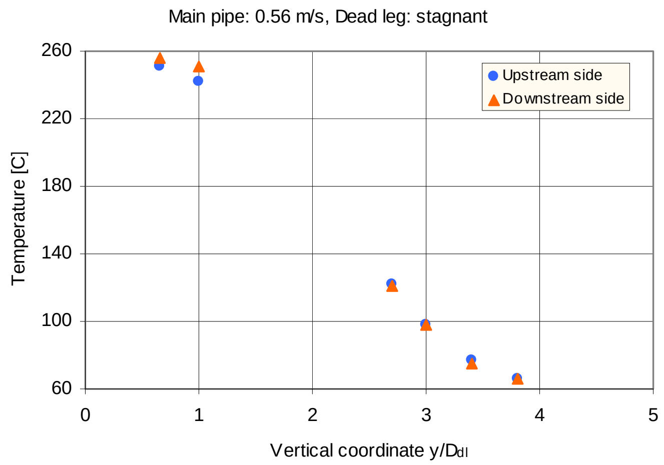

Figure 60. Inner wall temperatures of the dead leg in thermal stratification tests.

when the main pipe flow is 0.56 m/s and the fluid in the dead leg is stagnant. An exact value for the length of the vortex can not be given because of lacking thermocouples at the location of the flanges in the dead leg. At y/D_{dl} = 2.7, the temperature is practically constant across the dead leg indicating that the vortex has died away. Thus, the length of the vortex is estimated to be y/D_{dl} = 2.0–2.5. The steepest change in wall temperature occurs close to the lower end of the vortex in the transient and laminar region, where the heat transport decreases drastically. The situation does not change much, if the velocity in the main pipe is smaller (0.27 m/s). The maximum temperature difference between the downstream and upstream side of the dead leg is about 10 ºC, the same as in the high velocity case. Only the length of the vortex is now shorter.

With a small outflow (0.9 mm/s, about 6.2 litres per hour) from the lower end of the dead leg, the results are qualitatively different from the stagnant flow situation. Fluid at a temperature between 100–200 ºC occupies the upper half of the dead leg. As the hot water flows downwards in the uninsulated dead leg, it cools down by heat convection and radiation into the surrounding room. As a result, the temperature gradient becomes smoother and the stratification front stretches.

CFD simulations

The thermal stratification experiments described above were modelled with CFD calculations. The flow equations were solved with the commercial Fluent 5.5 code [4]. The transition from the turbulent to laminar flow was modelled by using the combination of the standard k-ε and the two-layer zonal (TLZ) models. For comparison, calculations with the Yang and Shih turbulence model were also performed.

The flow velocity in the hot (250 °C) main pipe was v_{main} = 60 cm/s, which corresponds to a Reynolds number of Re = 1.1×10^6. Therefore, the flow was fully turbulent. In the dead leg, the flow velocity was very small, and therefore the flow was laminar. We compare thermal stratification in two different situations. First, the flow at the lower end of the dead leg is stagnant (v_{dl} = 0). Second, the effect of a small outflow in the dead leg is considered (v_{dl} = –1 mm/s). A few additional cases are discussed in Ref. [5].

The shear stress at the mouth of the dead leg drives a vortex in the dead leg, see Figure 61. The effect of the small outflow of water in the dead leg is illustrated in Figure 61(b). The leak in the dead leg is found to increase the size of vortex somewhat.

In Figure 62, the temperature in the dead leg is presented. Thermal stratification is found to occur below the vortices, where heat transport decreases drastically. In the laminar region, the temperature approaches slowly the room temperature.

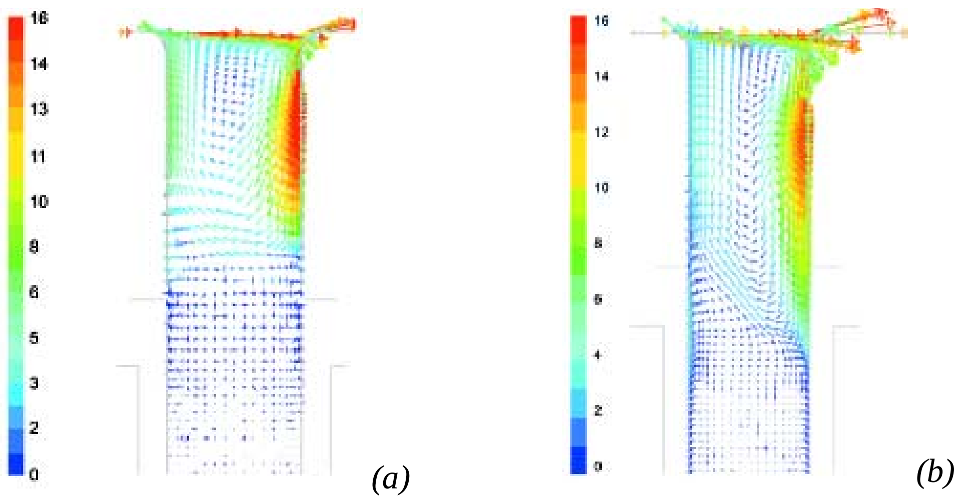

Figure 61. Velocity vectors coloured by velocity magnitude (cm/s). Vertical cross-section is shown, when the average flow velocity in the dead leg is (a) v_{dl} = 0, and (b) v_{dl} = –1 mm/s (v_{main} = 60 cm/s).

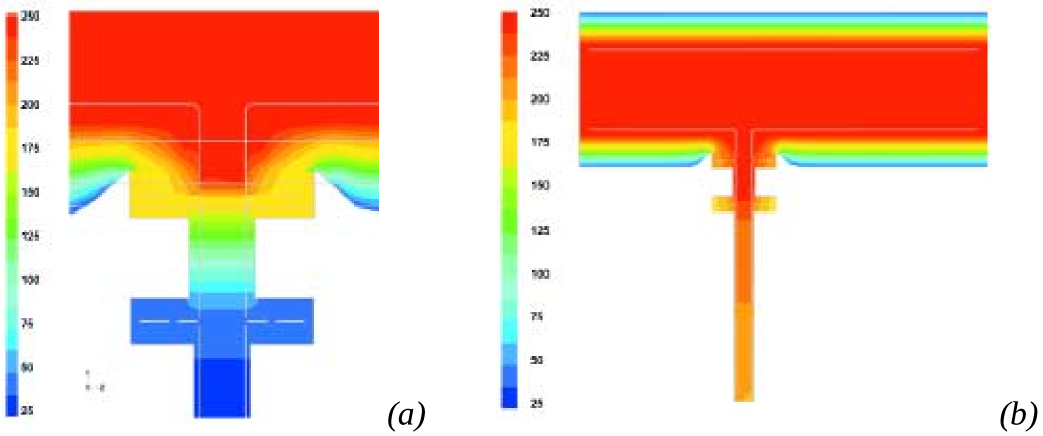

Figure 62. Temperature (°C) in the dead leg. The average flow velocity in the dead leg is (a) v_{dl} = 0 and (b) v_{dl} = –1 mm/s (v_{main} = 60 cm/s).

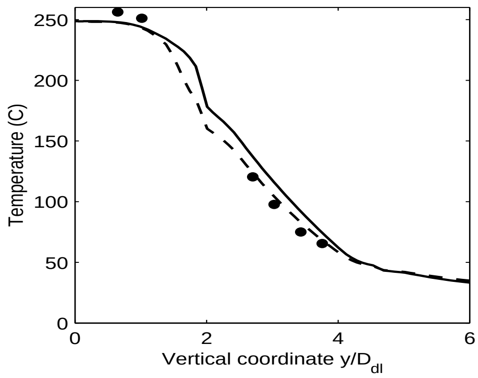

Figure 63. Simulation results obtained with the Yang and Shih model (dashed line) and with the standard k-ε and TLZ models (solid line) are compared with the experimental results (black circles). Near-wall temperature along a vertical line on the downstream side is shown (v_{main} = 60 cm/s, v_{dl} = 0).

In Figure 62(b), the effect of a small outflow of water from the dead leg is illustrated (v_{dl} = –1 mm/s). Heat is convected downwards from the main pipe by the small downward flow until the dead leg is filled with hot water. It has also been shown that a small inflow of cold water has a significant effect on the thermally stratified layer [5].

In Figure 63, the simulations are compared to the experimental results. The temperatures are shown near the wall on a vertical line on the downstream side of the dead leg. The experiment shown in Figure 63 has not fully achieved the steady state, and the measured temperatures were still slowly rising. It is, however, evident that the position of the stratified layer is calculated with a reasonable accuracy.

The results above show that a periodically leaking valve can cause significant cyclic heat loads on the wall of the dead leg. A periodic outflow of water can cause oscillation between the temperature distributions shown in Figures 62(a) and (b). In such an oscillation, the change of temperature is large enough to be of interest from the point of view of structural integrity.

Structural analyses

The structural analyses were mainly carried out by using the ABAQUS finite element code [6]. In addition, the test section was modelled and analysed separately with an axi-symmetric model that used the measured temperatures as a heat transfer boundary condition. The FE model was verified by comparing the simulated strains to the measured ones. The experimental measurements were for their part verified with simple analytical calculations of ideal cases.

The heat loads on the inner pipe walls were obtained from the CFD calculations. The inner wall geometry of the pipes used in the CFD calculations was imported to the FEM pre-processing programme, where the material properties and mechanical boundary conditions were assigned to the part and the calculation mesh was created. The calculated inner wall temperatures were transferred and interpolated from the CFD mesh to the structural analysis mesh with the help of an interpolation program.

Only a short section of the test facility is modelled and the mechanical boundary conditions and thermal boundary conditions on the outer wall are approximated. The inner pressure in the whole model is 70 bar (7 MPa). The number of shell elements in the main model is 1472. The element mesh is partly shown in Figure 64. The material properties are based on literature. The heat transfer and the static stress analyses are solved separately. Two cases are solved, where the temperatures of the two different quasi-stationary states of the CFD analyses are used as heat transfer boundary conditions on the inner wall. In the first case, the flow at the lower end of the dead leg is stagnant. In the second one, the effect of a small outflow in the dead leg is considered.

The values in the outer wall calculated with FEM are approximately two degrees lower than in the inner wall, which corresponds to the temperature measurements and verifies the heat transfer analyses. The strains develop due to the inner pressure and thermal expansion. Axial stress distributions (σ_{22}) on the inner surface in the above-mentioned cases are shown in Figure 64. The attention is paid now to the changeable test section mounted with two flanges. Positive stress values are tension (red colour) and negative values are compression (blue colour). Deformations in the figure are multiplied to make the image more illustrative. The temperature gradient is clearly steeper in the first case and the shape distortion due to thermal expansion is more evident.

In the first case, the test section is trying to expand due to rising temperature, but the upper flange is counteracting this expansion. The peak axial stress values in the test

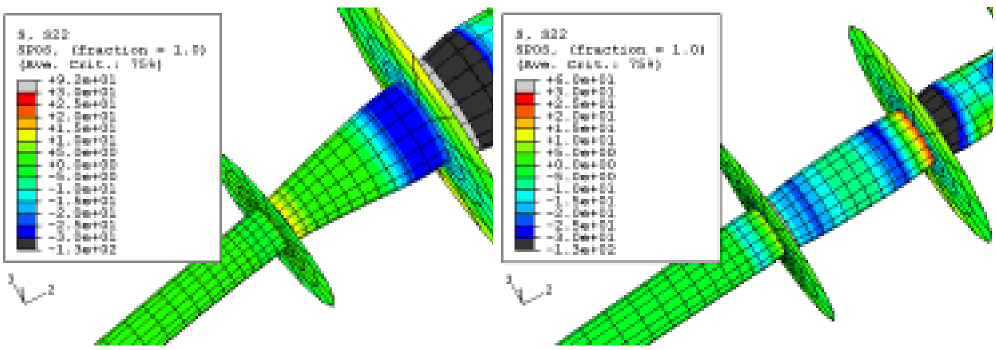

Figure 64. Axial stress distribution on the inner wall. The first case is on the left, the second one on the right.

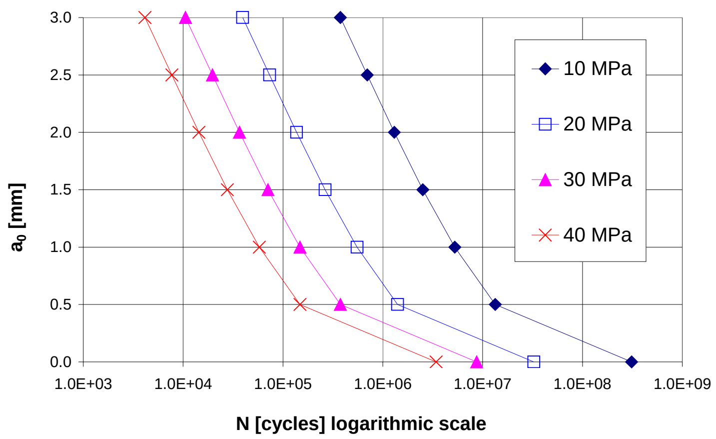

Figure 65. Amount of cycles to the rupture with different initial crack sizes and stress amplitudes.

section rise clearly above 20 MPa in magnitude, but in the centre of the section, the values stay between -10 MPa and 10 MPa. In the second case, where the dead leg is filled with hot water, both flanges are trying to prevent the test section from expanding. In the centre, the values stay approximately between -20 MPa and 20 MPa. In both cases, the axial stresses in the inner and outer wall are of opposite sign [7].

The reason behind the examination of these two different cases is to numerically simulate the effect of a periodically leaking valve. The stress states in the two extreme situations have been defined. The axial stresses at some locations keep cycling with amplitude of approximately 30 MPa. The peak von Mises stresses are approximately 80 MPa in the joint area. The material does not have any plastic deformations at any moment of the cycle.

If the stress and material toughness is known, fracture mechanics relationships can predict the critical flaw size in the structure. The cracks in the inner surface of the pipes are most critical when considering the fatigue behaviour. The stress intensity factor is solved using the linear elastic equations for the axially symmetric crack on the inside of a hollow cylinder. The range of variation of this factor is placed in Paris crack growth law. The number of cycles needed to reach each increment is numerically integrated. If a cycle duration of 10 minutes is assumed, one million cycles last approximately 19 years. The amount of cycles to the rupture with different initial crack sizes and stress amplitudes are shown in Figure 65 [7].

Conclusions

In co-operation between two FINNUS projects (TOKE and STIN), heat loads resulting from thermal stratification in a piping system dead leg has been studied. Experiments have been performed with a test loop built for this purpose. Local loading effects due to water flow and thermal stratification in a T-joint of a hot horizontal pipe and a cold vertical dead leg were measured with densely positioned thermocouples. The response of the pipes to the thermal loads was measured with strain gauges.

The experiment was modelled numerically with CFD simulations and structural analyses. The CFD calculations show that a vortex forms in the top part of the dead leg as a result of hot water flowing into the dead leg from the main pipe. Comparison with experiments shows that the position and the width of the thermally stratified layer could be calculated with a reasonable accuracy.

The heat loads found in the CFD calculations were successfully transferred to the structural analyses. The FE results were indicative and qualitatively realistic. As a result of these projects, a tool for more specific assessment of thermal stratification in a T-joint of pipes has been created.

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10. Modelling and simulant experiments of severe accident phenomena (MOSES)

10.1 MOSES summary report

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Abstract

MOSES project has systematically investigated phases of severe reactor accidents relevant to Finnish nuclear power plants. The specific areas of research have been pressure vessel failure mode, core debris coolability, fission product behavior, including chemistry, containment thermal hydraulic loading, especially hydrogen detonations and phenomena relevant for long-term severe accident management. The outcome of the project has been novel analysis tools, procedures and tools to perform multi-diciplinary analyses and new experimental results reducing noticeably uncertainties that existed in the field of severe accidents.

Introduction

The safety of the Finnish nuclear power plants against severe reactor 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 areas are coolability of core debris, failure mode of the pressure vessel and fission product behaviour, particularly in the reactor system. In order to address the identified research needs, the project has focussed 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

The project has systematically addressed the phases of severe reactor accidents relevant for the Finnish nuclear power plants. Olkiluoto BWR was selected as a reference plant for assessments where plant specific parameters or data were needed. More importantly, the approach in investigations of different physical phenomena is generic making the results applicable also to Loviisa plants. The MOSES project has focused on investigations of five specific areas connected to the progress and development of a postulated severe accident in Light Water Reactors. The project has applied different technical methods in tackling the various problems ranging from rough hand calculations to complex 3-

dimensional simulations and on the other hand from critical follow-up and participation in international experimental research to performance of own, simulant experiments tailored to address features of Finnish reactors.

The key areas of research in MOSES project are:

- Investigate pressure vessel lower head failure mode under severe accident conditions
- Evaluate core debris coolability in the containment
- Examine behaviour and chemistry of fission products
- Assess the phenomena contributing to containment loading in severe accidents (hydrogen combustion, detonation mechanisms)
- Investigate the most important phenomena considering severe accident management in long term.

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 1/5th 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 studied coolability of granular debris by measuring dryout heat flux in simulant material particle bed. The particle bed was designed to be representative of the situation in Olkiluoto pedestal during a severe accident. The key parameters were the particle size distribution and the bed depth. The experiments will define the dryout heat flux of the particle bed, i.e. the minimum heating power in the bed, that will form a local stable dry zone some where in the bed. The dryout heat flux also sets an upper limit to the decay heat per unit area that can be extracted from the melt by water penetrating the bed from above. 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 fourth major target area is the assessment of hydrogen detonations in a BWR reactor building. The task comprised 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 were performed. The work on hydrogen detonations was also part of Nordic SOS-2.3 project. Finally, the last task is to investigate the long-term severe accident management in respect of anticipated radioactivity levels and cumulative doses in the containment and the chemical aspects like radiolysis of water producing extra hydrogen into the containment and the solubility of uranium into containment water pool under circumstances of pH control.

Pressure vessel lower head integrity

The work on pressure vessel failure mode focused on follow-up Sandia OLHF-creep rupture experiments and the simulation of the tests with the FEM-model developed at VTT [1]. 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 50–125 bars and simultaneously heated uniformly. A total of four tests were performed with varying pressurisation strategy and for lower heads with and without penetrations. The test programme produced an extensive and well-documented database on material properties and measured temperature and deformation maps for development and validation of calculational models as well as for increasing the understanding of creep rupture phenomenology. Two tests were selected for calculational analysis at VTT Processes: OLHF-1 without penetrations and OLHF-4 with four lower head penetrations. The simulations were performed with both 2-dimensional and 3-dimensional models. Both models were able to predict the behaviour of the lower head structure excellently for both lower head types [2], (see also a special report below). The failure of the pressure vessel occurred at the bottom of the vessel in OLHF-1 test and at the interface joining the penetration weld to the vessel wall in the OLHF-4 test. The creep failure occurs at a location, where the temperature is highest and if the temperature profile is uniform, at the location, where the vessel wall is thinnest.

Melt coolability

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 [3]. 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 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.

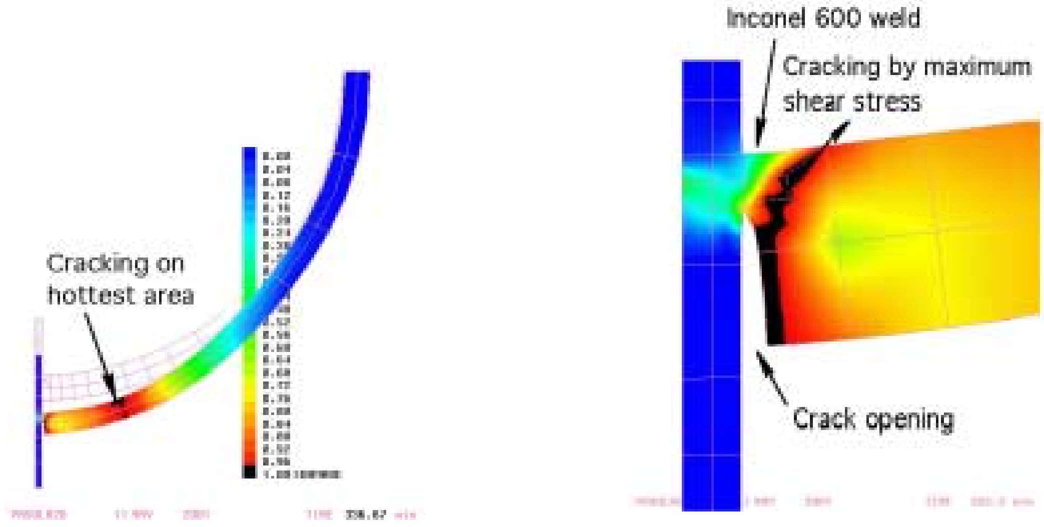

Figure 66. Example of PASULA application on analysis of lower head failure. Penetration probably fails before the vessel wall only if the penetration is in the hottest area. Inconel creeps less than base material.

The dry bed heat transfer experiments were calculated with the PASULA and a specially developed BEDEXP codes [4], [5]. 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. 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.

The problem of particle bed coolability has been approached first by defining a representative particle size distribution for Olkiluoto plant [6]. The mass-averaged particle size of this distribution was determined to be 3.46 mm with the particle size ranging from 0.25 mm to 11 mm. Second, a literature review on existing debris bed coolability experiments was carried out [7]. An extensive dryout heat flux database exists for beds of single-sized, spherical particles. Some data is available for homogeneously mixed or stratified beds with particles having a narrow size distribution (POMECO tests), and for homogeneous and stratified beds with wider particle size range (DCC tests). In general, the measured data showed large scattering and none of the available experiments fitted exactly to the particle size distribution or bed arrangement expected in Olkiluoto plant. Furthermore, the available calculational models were very sensitive to the particle size and bed porosity, yielding results from easily coolable to non-coolable situation with a relatively small change in parameter values. Most of the measured data and all the calculational models suggested that a bed with average particle size of 3.5 mm would have a dryout heat flux close to 1 MW/m^2 and thus would be easily coolable.

For final check of the effects of Olkiluoto specific particle bed characteristics on dryout heat flux an own simulant material test facility, STYX-1 was constructed at VTT (see a special report below).

The simulated debris bed consists of alumina particles with special size distribution immersed in water. The mass averaged particle diameter of this “sand” is 3.37 mm, the surface averaged diameter is 1.87 mm and the average diameter based on the number of particles is 0.72 mm. The bed material was prepared in small batches to ensure uniform distribution throughout the bed. The shape of the bed is cylindrical with diameter of 300 mm and depth of 600 mm and it is placed in container cylinder inside a pressure vessel. Both the inner bed container and the pressure vessel are made of stainless steel, and the maximum pressure inside the vessel is 7 bar (6 bar above the atmospheric pressure).

The water level is set to approximately 175 mm above the bed surface and it is measured during the test. The water used is purified with a conductivity of 15–20 µS. When needed, water is added from the top of the bed by feedwater valve and the steam generated in the bed is led out through pressure valve controlled by a pressure gauge. There is also another pressure gauge that is used for measurement. The test bed is heated by cylindrically shaped three-phase resistance wire element distributed horizontally on six heating levels set to 50, 150, 250, 350, 450 and 550 mm from the bottom of the bed. The maximum reachable heating power is 87 kW.

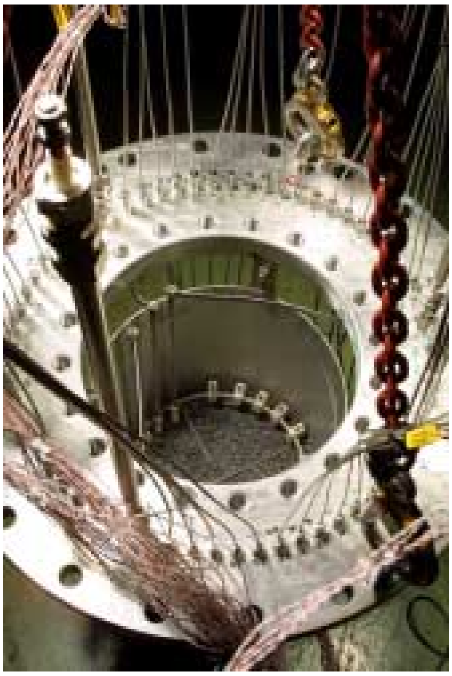

Figure 67. STYX-1 pressure vessel flange with thermocouple inlets. Particle bed visible inside the pressure vessel.

Figure 68. Inner cylinder of STYX-1 facility containing the particle bed with thermocouple penetrations through the cylinder.

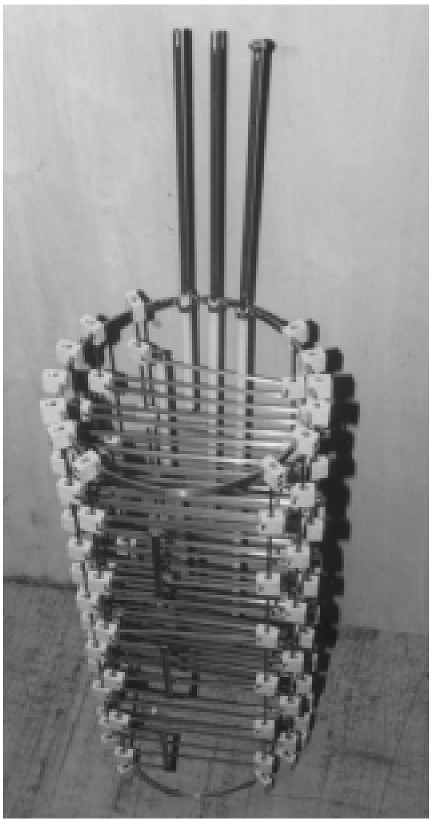

Figure 69. Heater element of STYX-1 test apparatus with 3-phase power input wires.

Figure 70. A top view into the test section. The topmost heater level and three thermocouples visible on top.

Temperature of the particle bed is measured by 52 thermocouples distributed in seven layers set to 10, 100, 200, 300, 400, 500 and 590 mm from the bottom of the bed, and an additional layer located 610 mm from the bottom. The uppermost thermocouples are normally above the particle bed and are for future stratified bed measurements, in which an additional thin layer of small-sized particles will be added on top of the bed. There are also some additional thermocouples located on the walls of the pressure vessel and inner container.

Two test series STYX-1 and STYX-2 have been performed with the uniformly mixed bed in pressures varying from atmospheric to 7 bar absolute pressure. The first test series ended into failure of the heater wire. After reaching the dryout near the bottom of the bed, the power was increased in attempt to investigate the expansion of the dry zone. During this phase the heater wire exceeded its maximum operation temperature. During the changing of the heater element, additional thermocouples were placed near the bottom of the bed and on the top of the bed. The dry zone formed in all tests with uniformly mixed bed near the bottom of the bed. Table 8 shows the key results of the dryout experiments.

Table 8. Key results from the dryout heat flux experiments with STYX-1 facility with uniformly mixed particle size distribution.

Test	Pressure [bar]	Measured dryout power [kW]	Corresponding dryout heat flux [kW/m^2]	Elevation, where dry zone was first detected	Comments
STYX-1/1	3	32	485	1 cm from the bottom	large (12 kW) power step to dryout power
STYX-1/2	3	26	388	1 cm from the bottom	power step 3 kW to dryout power
STYX-1/3	5	36	509	1cm from the bottom	failure of heater after reaching dryout
STYX-2/1	5	26	371	7 cm from the bottom	dryout detected with new thermocouples
STYX-2/2	7	32	451	3 cm from the bottom	dryout detected with new thermocouples
STYX-2/3	1.15	16	232	7 cm from the bottom	
STYX-2/4	2	21	300	4 cm from the bottom	

A third test series, STYX-3, will be performed during the fall 2002 investigating the dryout heat flux in a stratified bed. The current facility is modified by adding 6–8 cm thick layer of fine sand (grain size 0.25–0.4 mm) on top of the bed. This will simulate a situation, where fine fragments resulting from an energetic fuel-coolant interaction are deposited on top of the coarser bulk of the bed.

Containment integrity

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 [8]. 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.

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 [9]. 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 [10]. The detailed analyses yielded lower maximum detonation pressure of about 7 MPa, but longer static type pressure 'tail' of about 0.5 MPa accounting for the 3-D shock reflections and continuous combustion process. The maximum impulses predicted by DET3D code were higher, about 35 kPa-s. The pressure and impulse maxima concentrated to the corners of the room.

A computer code interface was 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 were calculated with DET3D code [11]. The walls of the reactor building room may survive the peak detonation, but the relatively slowly decreasing static type pressure following a peak detonation damages the wall more severely. The integrity of the pipe penetration under these circumstances was analysed with the ABAQUS code in the STIN project. Even with conservative assumptions, the plastic deformations caused to the pipeline penetration during a detonation would be 3.5 % at maximum. This is well below the success criteria found in the literature. This work has been performed as part of the Nordic nuclear research programme NKS/SOS-2.

Fission product behaviour

Another topic, that has been worked upon in Nordic collaboration, is the behaviour of organic iodine. The Nordic project, co-ordinated by VTT Processes, 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 [12]. 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, such as silver and zinc.

Long-term accident management

Severe accident management is primarily focused on mitigating the consequences of severe accidents and reaching a stable state of the plant. Long-term accident management can be considered to cover the period from a few days to several years. The main goal in the long-term actions is to maintain the plant in a stable state and prevent any uncontrolled releases of radioactivity to environment.

The major concern in severe accidents is caused by the radioactivity of fuel materials released from the damaged core. High radioactivity level may have a direct impact on materials of components, barriers and safety-related equipment. Radioactivity may also affect many chemical processes, like water radiolysis and iodine chemistry inside the containment atmosphere, water pools and primary circuit [13]. Extremely harmful consequence in respect of long-term accident management is that the radioactivity level may prevent partly or totally the access to the areas essential for maintenance and recovery actions. This may impede the accident management efforts severely. Therefore, it is apparent that an early characterisation of the radiological conditions of the containment (and reactor building) is necessary to provide data e.g. for strategic planning of accident management and other recovery actions.

A sample accident scenario was calculated with MELCOR code for Olkiluoto plant for definition of fission product transport in the containment following a core melt accident. The radioactivities and decay processes were calculated separately with ORIGEN2 code. The results were combined to estimate the total radioactivity level in different compartments of the containment. The accident case was analysed for both assuming the containment pressure control via operation one loop of pool cooling and containment spray (no venting) and by assuming initial failure of pool cooling system leading to pressure relief once through filtered venting system (venting), after which one loop of pool cooling system was assumed to be recovered. Figure 71 shows the radioactivity in the drywell in both analysed cases.

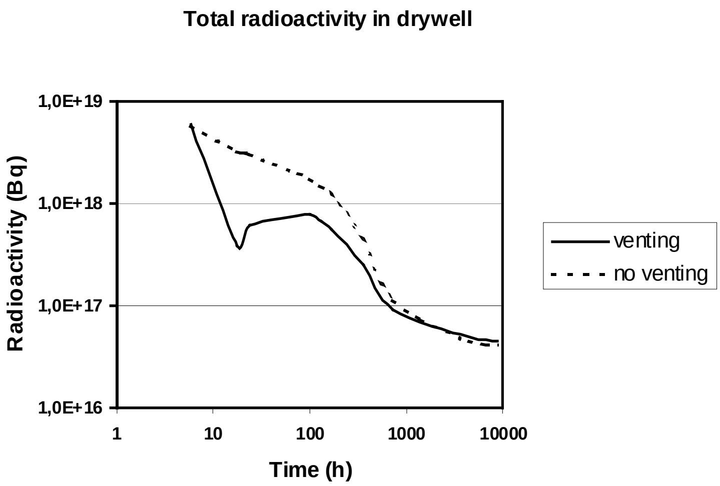

Figure 71. Radioactivity in the drywell of Olkiluoto reactor following a hypothetical severe accident.

Irradiation of water in a contaminated containment pool leads to build-up of a steady-state concentration of hydrogen peroxide in solution and the continual escape of hydrogen and oxygen from the system. The first estimate of hydrogen production from water radiolysis is about 1000-kg/a [13].

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. The obtained results can be used to assess the coolability of the granular particle bed in Olkiluoto pedestal in a geometry, where debris is assumed to spread uniformly on the whole pedestal floor. This debris geometry, though a conservative one, is a good basis for design of coolability concepts. The performed test programme indicated the significant effect of wide particle size range in the debris and the irregular shape of the particles in reducing the dryout heat fluxes in comparison to the measured values for respective beds of more uniform-sized, spherical particles. With supportive calculations an assessment can be performed, if a particle bed is coolable on the containment floor with the current accident management measures. However, international research and experiments are needed to evaluate a melt pool coolability from the top. The answers to melt coolability were searched from MACE /ACEX tests and currently from OECD/MCCI tests. These test programmes have been followed-up as part of MOSES-project.

The assessment of hydrogen detonation issue in the reactor building yielded practical conclusions for Olkiluoto plant. The detailed detonation load simulations combined with the analysis of structural behaviour of the walls of a sample Olkiluoto reactor building room B.60.80 and the system 321 (shutdown cooling system) pipeline penetration. The analyses showed that the concrete walls of the room may loose leak-tightness, but the more importantly, the pipe penetration would survive the detonation intact. This analysis effort created a procedure for combining detailed load analyses for rapid, energetic combustion phenomena and assessments of behaviour of structures exposed to these loads. The analysis procedure is applicable also to non-nuclear problems.

The long-term accident management evaluation task produces information of radioactivity levels in various parts of the containment building, needed for assessment of human accessibility of containment for maintenance operations as well as for the assessment the survivability of devices in the radiation field. Also the long-term phenomena, such as slow hydrogen production from water radiolysis and the possible extra contamination factor by solution of uranium from debris bed into containment water pool is needed information for stabilisation of severely damaged reactor and accident management in several years perspective. The long-term accident management issues have received significantly less attention also internationally than the short-term accident management issues. However, the examples of TMI-2 and Chernobyl accidents have revealed the complexity of the reactor stabilisation, cleanup and long-term accident management following a severe accident.

Conclusions

MOSES project has produced novel analysis tools, procedures to perform multi-diciplinary analyses, new experimental results in the areas where noticeable uncertainties existed in the severe accident field.

A new FEM-model was developed and validated against experiments for 2- and 3-dimensional analyses of pressure vessel lower head creep behaviour. The validation covered lower head geometries both with and without penetrations. The agreement of measured and calculated results was excellent. The PASULA/FEM model is applicable to both Finnish reactor types.

A capability of analysing consequences of rapid, energetic combustion has been developed in collaboration between MOSES and STIN projects. The analysis capabilities cover the 3-dimensional simulation of thermal-hydraulic loads related to super-sonic combustion with DET3D code developed at Forschungszentrum Karlsruhe, an interface code developed at VTT to generate loading sequences for structural analysis code ABAQUS and the simulation of concrete or steel structure behaviour under defined loads. The capability was used in evaluation, that of system 321-pipe penetration outside the containment most likely maintains leak-tightness during a hydrogen detonation in the surrounding reactor building in Olkiluoto plant.

Long-term sever accident management issues have been studied by evaluating the radiation levels in the containment atmosphere and water pools following a severe accident. The analysis accounts for fission product mass transport and radioactive decay processes. The results can be used to assess human accessibility into the containment and the survivability of various accident management devices in the high radiation field in the containment. The chemistry related consequences, such as hydrogen production from radiolysis of water and solubility of uranium into containment water pools have been assessed utilising the determined distribution of radioactivity.

The coolability of particulate core debris in the containment water pool has been assessed with own simulant experiments. The experiments measured so-called dryout heat flux, which is the maximum amount of heat per unit area that can be extracted from the core debris by water supplied from the top of bed. The experiments were designed to be representative of the situation in Olkiluoto pedestal during a hypothetical severe accident. The measured dryout heat fluxes were lower than generally reported in international literature. The reason for lower dryout heat fluxes was that the particle size distribution of the debris bed was wide and the particle shape was irregular, resulting in a bed that behaves as if it were made of much smaller particles. The mass-averaged particle size of the bed was 3.46 mm and the defined effective particle size was only

0.804 mm. The dryout heat fluxes were in general slightly higher than the decay heat production per unit area in Olkiluoto plant. The experiments continue with investigations of stratified bed geometry, where a 6–8 cm thick layer of finer particles is settled on top of a 60 cm deep coarser particle bed. This simulates a situation, where part of the debris is fragmented finely during a more energetic fuel-coolant interaction.

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10.2 Mechanical behaviour of reactor pressure vessel in severe accident

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Abstract

This article describes the main achievements in developing methodology for analysing mechanical behaviour of pressure vessels with and without penetrations at high temperatures related to accident scenarios. Validation and applications of the methodology are presented.

Introduction

The study focused on the mechanical behaviour and possible failure of the reactor pressure vessel in a severe accident. The pressure vessel failure mode is an important initial condition for a subsequent containment integrity analysis. It should also be investigated whether melt retention within the pressure vessel by external cooling is a viable accident management concept. Furthermore, the time between melt relocation and the lower head failure of the pressure vessel may be important for some scenarios and plants.

Calculation methodology and material modelling of RPV steels was developed. Large displacement and large strain finite element analyses were performed by the PASULA computing package, which has been developed at VTT for a variety of structural analysis and heat conduction calculations. The codes are used to support design and interpretation of experiments and to extend the results of experiments to plant scale. Efficient visualisation programmes have been developed to support the PASULA code for visualising deformations, stress and temperature fields etc.

The methodology and calculation process were validated by comparing the calculated results with measurements from experiments on small scale pressure vessels. An excellent agreement was obtained between the measured and calculated results concerning deformations and failure location.

Continuum mechanics for large deformation

Due to high nonlinearities and time dependent temperature distributions the calculation is advanced by taking finite time increments. The basic unknown are the incremental displacements of the nodes. Due to nonlinearities the displacements are determined by iteration. The stress and strain states are considered in integration points of finite elements. A fairly general procedure, independent of material behaviour, for calculating finite deformation and stress state was developed for a finite element analysis. In this procedure, the change of stress and strain state during a time increment is considered in a co-ordinate system aligned in the directions of principal stretches caused by deformation during the time increment (Figure 72). This has the advantage that originally a rectangular material element is still rectangular after the deformation, and is slightly distorted only at the intermediate points of the deformation increment. The details of the method is presented in [2].

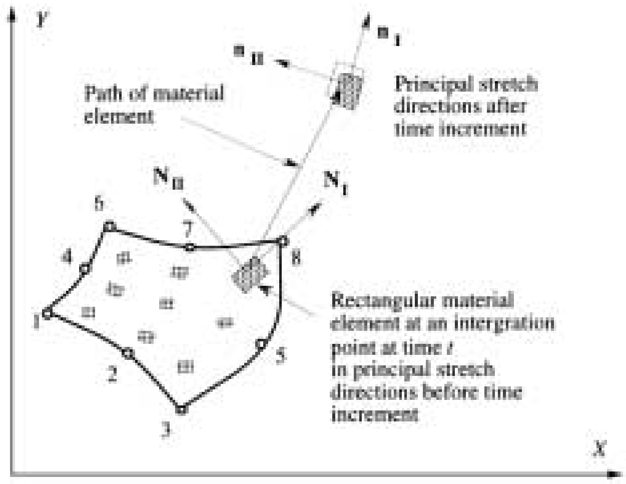


Figure 72. Illustration of principal stretch directions in a 2D case.

Constitutive modelling of reactor pressure vessel steels

It was found that uniaxial tensile, relaxation and creep tests within the temperature and time range of interest can be very satisfactorily described by a single inelastic deformation rate model of strain hardening type ($d\varepsilon_{ie}/dt$ is inelastic strain rate)

$$\frac{d\varepsilon_{e}}{dt} = A_T\, \varepsilon_{ie}^{s} \left(\frac{\sigma}{\sigma_{ref}} \right)^{r} \quad . \tag{13}$$

The model (13) is applied for describing both axial tensile, axial relaxation and axial creep experiments. The coefficient A_T depends strongly on temperature. The parameter s takes strain hardening into account (a typical value $s \approx 0$–0.5 for steels). Reference stress σ_{ref} is used to make the term inside parenthesis dimensionless. The parameter r (a typical value $r \approx 3.4$–4.0 for steels) is related to the stress. The parameters r and s are rather independent of the temperature. True stress and natural strain (logarithmic) values need to be used in creep modelling, since these quantities are consistent with stress and strain measures used in large strain analysis. There is no specific separation of plasticity and creep, since they are both caused by similar physical processes. Figure 73 shows the values of the parameter A_T for SA533B1 and for Inconel 600 [1], which is used as penetration nozzle and weld material in actual reactors. The modelling was extended to three-dimensional cases by applying the von Mises hypothesis for calculating effective stress.

The basic model (13) does not follow the measured curve at greater strain than about 0.1 at low temperatures in the ferritic phase. That is why an additional term was introduced into the model as follows:

$$
\begin{aligned}
\frac{d\varepsilon_{ie}}{dt} &= A_T \varepsilon_{ie}^{s} \left(\frac{\sigma}{\sigma_{ref}}\right)^r & \varepsilon_{ie} < \varepsilon_a \\
\frac{d\varepsilon_{ie}}{dt} &= A_T \varepsilon_{a}^{s} \left[1 + A_a(\varepsilon_{ie} - \varepsilon_a)\right] \left(\frac{\sigma}{\sigma_{ref}}\right)^r & \varepsilon_{ie} > \varepsilon_a,
\end{aligned}
\tag{14}
$$

where ε_a is the threshold value and A_a is a coefficient. Figure 74 shows the parameters. In austenitic phase the additional term is not necessary. In reactor cases small inelastic deformations (<0.01) take place at low temperatures. Large deformations necessary for failure take place at high temperatures (>900°C) in the austenitic phase, where the strain hardening model works well. The model parameters demonstrate that reactor pressure vessel steels behave in quite a different way in ferritic and austenitic phases.

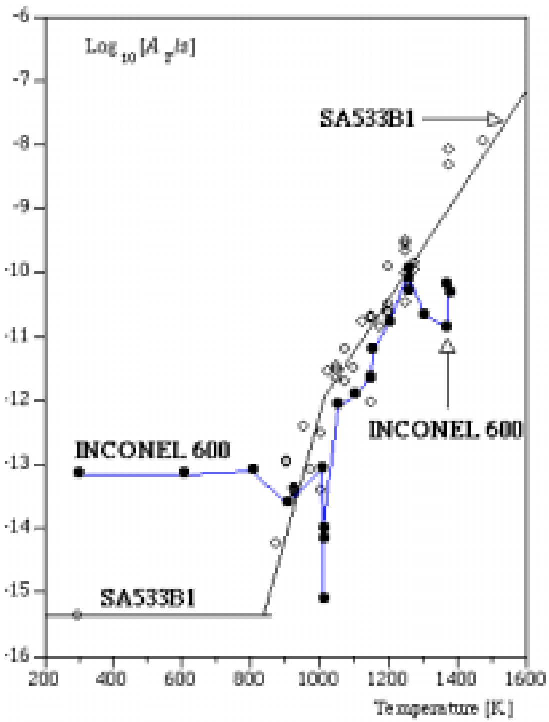


Figure 73. Parameter A_T determined from tensile and creep measurements of the RPV steel SA533B1 and Inconel 600.

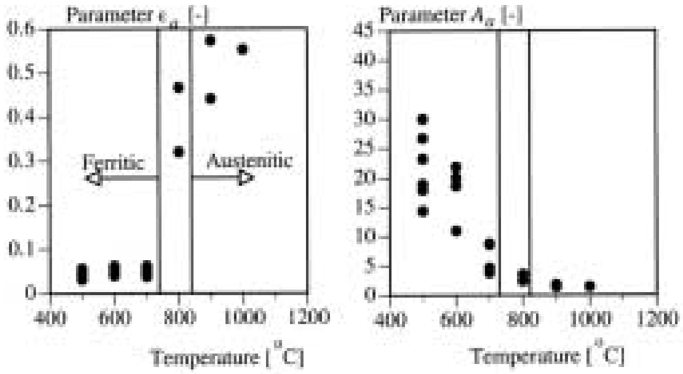


Figure 74. Parameters ε_a and A_a as a function of temperature.

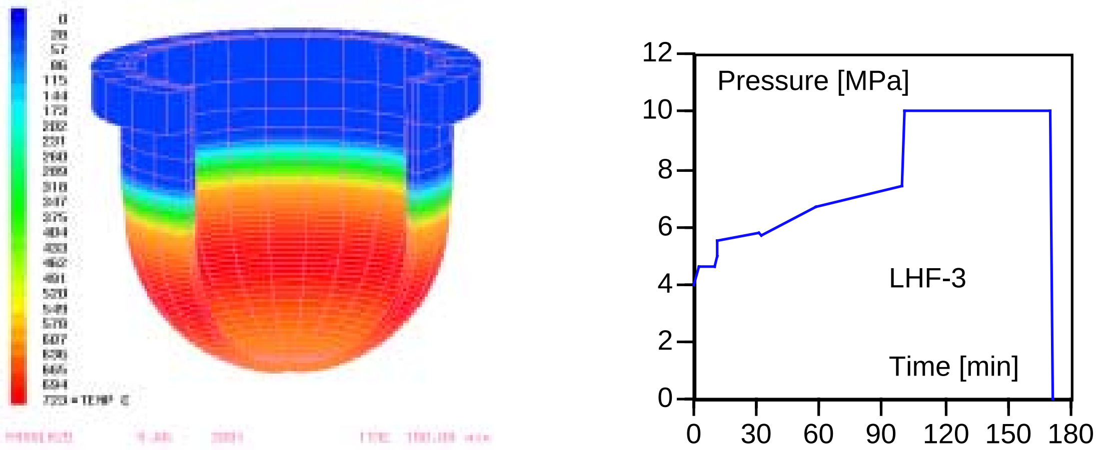
Evaluation of failure

At high temperatures RPV steel behaves in a ductile way. Most of the strains before fracture are caused by primary stresses due to the pressure load. Primary stresses, which do not relax, cause essentially a two-dimensional stress state in a cylindrical or spherical reactor pressure vessel. Secondary stresses caused by thermal gradients relax at high temperatures. Secondary stresses cause fairly small strains compared with the strains needed for a fracture at high temperatures. The ratio of the hydrostatic stress to the von Mises equivalent stress is a versatile quantity for characterising the triaxiality of the stress state. In a spherical RPV and in the membrane stress state caused by a pressure load, the stress state characterising triaxiality factor is 2/3 = 0.666. In a cylindrical pressurised RPV the triaxiality factor $1/\sqrt{3} \approx 0.577$ is of same magnitude (in uniaxial testing the triaxiality factor is 1/3 when neglecting the triaxiality in the necking area).

Natural strain ε_{ie} in Equation (13) can be used as a failure measure for estimating rupture. The cumulative strain can be thought to describe the internal damage of the material. The critical value of the cumulative strain was determined by simulating small scale RPV experiments by FEM. The critical strain depends on the sulphur content (MnS_2) in the metal. A review of experiments done for small-scale pressure vessels showed that failure takes place when the effective cumulative inelastic strain reaches a critical value of about 0.3 in case of high sulphur content and about 0.6 in case of low sulphur content.

Validation and application examples of the calculation process

Validation of the calculation process concerning e.g. capabilities for large inelastic deformation analysis is very important for ensuring that the models describe the real behaviour of reactor pressure vessel. A reliable validation can only be based on experiments in order to demonstrate that the models describe real physics to a sufficient level of accuracy.

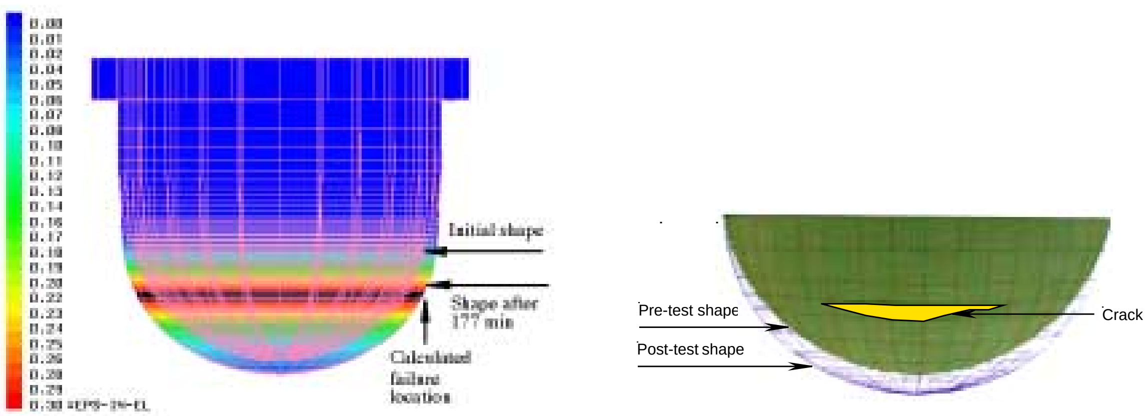


(a) (b)

Figure 75. Steady state temperature distribution (a) and pressure load history (b).

Small-scale experiments conducted and reported in international projects are the most important type of validation. The LHF experimental programme conducted by Sandia National Laboratories consisted of eight lower head failure tests, from LHF-1 to LHF-8 [4]. Hemispherical lower heads, similar in shape to the TMI-2 reactor pressure vessel, were subjected to internal pressure and thermal loading. The temperature was relatively low and the pressure relatively high compared to reactor cases (Figure 75). The temperatures are below the phase transformation zone. The diameter of the lower head was D_i = 0.91 m corresponding to a geometrical scale factor of 4.85. The pressure vessels of the tests were made of steel SA533B1.

The effective cumulative strain reached its maximum in the hottest area and was 0.31 when the LHF-3 vessel failed during the test after 177 minutes. Figure 76a shows the inelastic strain distribution. At the time of failure the maximum strain was close to the horizontal crack location, as shown in the post-test photograph in Figure 76b.



(a) (b)

Figure 76. Inelastic effective cumulative strain distribution after 177 min (a) and post-test photograph for LHF-3 [5] (b).

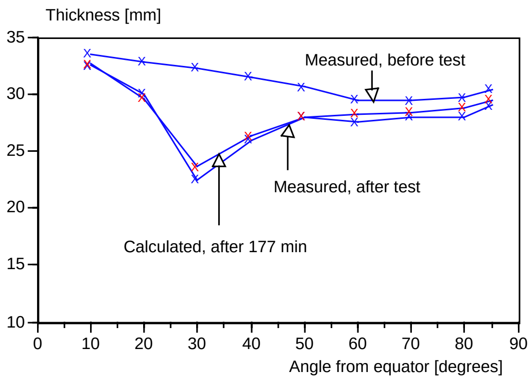


Figure 77. Measured and calculated wall thickness profiles in LHF-3 test.

Pre-test and post-test measured [5] and calculated [3] thickness profiles are shown in Figure 77. Agreement between the measured and calculated thickness is satisfactory.

Figure 78 demonstrates the results of three-dimensional analyses, which were performed for pressure vessels with and without penetration. A rather coarse element mesh can give nearly equal deformations and strain distributions as a more dense mesh, since at high temperature the secondary stress peaks are relaxed and the stress state is rather smooth. The measured variable pre-test wall thickness was taken into account in generation of the element mesh. The load vector due to measured pressure loads acting on the internal surfaces and the gravity load of the wall were updated continuously with the deformed mesh. Measured temperatures on internal and external surfaces were spatially interpolated to the nodes at different times. Deformations and failure location were well predicted when compared to measured values.

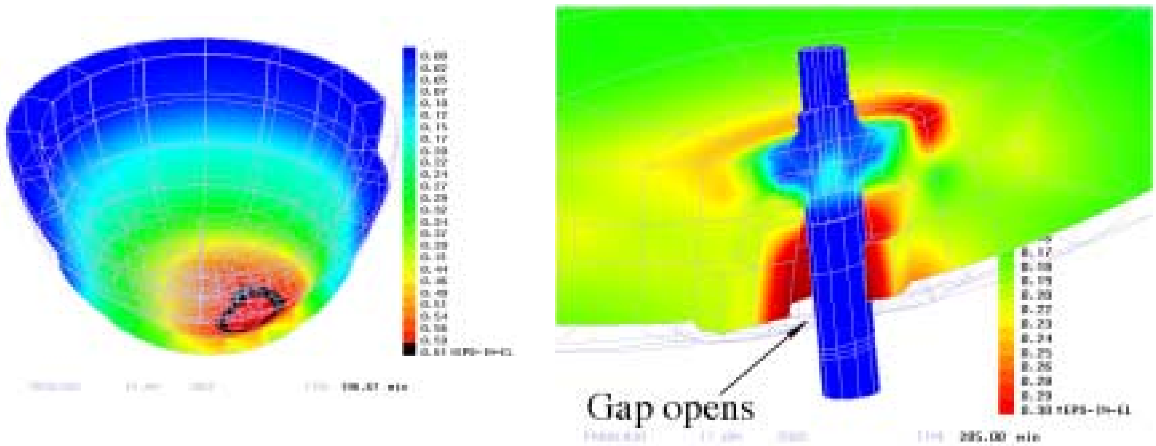


Figure 78. Calculated deformations and inelastic strain distribution of spherical vessel bottom without penetration (a) and with penetration (b) by 3D FEM model .

The penetrations for instrumentation and control rods through the lower plenum of a reactor pressure vessel wall may be the weak points in core melt accidents. Depending e.g. on the locations of welds, vessel failure may initiate at the penetration. A penetration disturbs the global stress field in the vessel wall only over a small area. Two penetrations interact only if they are very close to each other. That is why a single penetration can be analysed separately. A detailed analysis of a penetration requires quite complex three-dimensional modelling. A special mesh generator, which optimises the mesh, was developed. The boundary between the base material and weld material does not follow element edges, but the material type in the model is determined according to the location of an integration (Gauss) point. This also reduces the number of elements and computing time.

Figure 78b shows the calculated strain distribution near the penetration. The initial cylindrical gap opens between the penetration tube and the vessel wall during the loading. Non-axisymmetric crack opening is obtained. The penetration rotates from its original vertical orientation. Figure 78b shows that the penetration rod and weld of Inconel 600 is creeping less than RPV steel. The weakest area is the heat affected zone between the weld and the base material. Typically, a crack is initiated on this zone.

In connection of 3D analysis the simulation of crack initiation and growth has been developed. In this method the through-thickness crack opening is increased according to local failure of the vessel wall. In case of ideal gas content the pressure inside the vessel is decreased according to the leak flow calculated from adiabatic equation for ideal gases.

Conclusion and discussion

Efficient computing capabilities for analysing pressure vessel behaviour in severe reactor accident has been developed. Validation cases show that the models can simulate well the experiments conducted by small scale vessels with and without penetrations and an excellent agreement was obtained between the measured and calculated results concerning deformation and failure location. The work increased remarkably knowledge and understanding of pressure vessel behaviour during severe reactor accident with and without penetrations.

The validation has shown that a failure can be predicted with reasonable accuracy by applying a strain based criterion. Most of the strains before failure are caused by primary stresses due to the pressure load. Primary stresses, which do not relax cause essentially a two-dimensional stress state. Critical effective strain of magnitude 0.3–0.6 depends on the sulphur content in the metal.

A penetration disturbs the global stress field in the vessel wall only over a small area. Two penetrations interact only if they are very close to each other. That is why a single penetration can be analysed separately. A single penetration has been successfully analysed by two and also three dimensional FEM models.

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10.3 Experiments on dryout heatflux in volumetrically heated granular particle bed with the STYX facility

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I. Lindholm, J. Miettinen and R. Sairanen, VTT Processes
H. Sjövall, Teollisuuden Voima Oy

Introduction

Core debris coolability in the containment is studied to verify the severe accident management strategy adapted in Olkiluoto BWRs. The molten core material discharged from the failed reactor pressure vessel falls into a several meters deep water pool in the Olkiluoto containment. It is most likely, that the melt will fragment on its way down in the subcooled water pool and form a self-heating particle debris bed on the pedestal floor. The coolability of this particle bed is the key question to be answered.

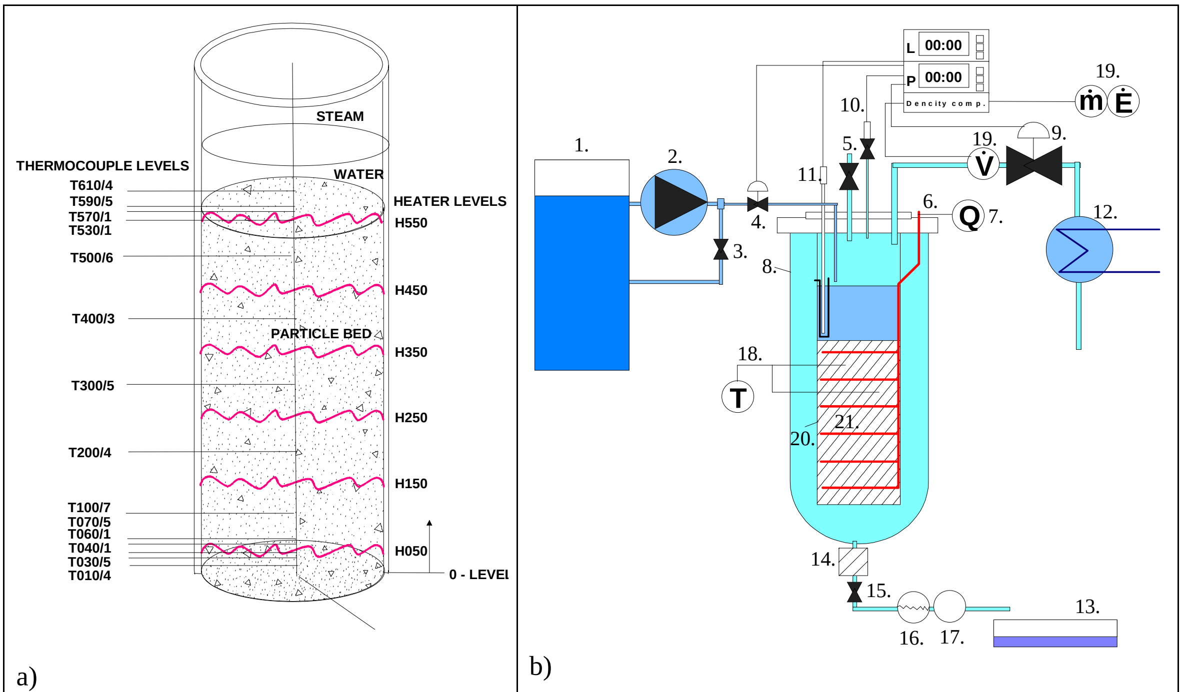
The problem of particle bed coolability has been approached first by defining the representative particle size distribution for Olkiluoto plant [1]. The mass-averaged particle size of this distribution was determined to be 3.46 mm with the particle size ranging from 0.25 mm to 11 mm. The particles formed in melt-coolant interaction tests were generally non-spherical. Second, a literature review on existing debris bed coolability experiments was carried out [3]. An extensive dryout heat flux database exists for beds of single-sized, spherical particles. Some data is available for homogeneously mixed or stratified beds with particles having a narrow size distribution from POMECO tests [4], and for homogeneous and stratified beds with wider particle size range from DCC tests [5],[6]. However, the particle shape may have been more uniform in the POMECO tests (natural sand) and DCC tests (UO_2 particles) than resulted e.g. FARO experiments and MACE tests with formation of particulate. The available calculational models were sensitive to the particle size and bed porosity, yielding results from easily coolable to non-coolable situation with a relatively small parameter range. Most of the measured data and all the calculational models suggested that a bed with average particle size of 3.5 mm would have a dryout heat flux close to 1 MW/m^2 and thus would be easily coolable.

For investigating of the effects of Olkiluoto specific particle bed characteristics on dryout heat flux an own simulant material test facility, STYX-1 [2] was constructed at VTT as a combined effort between VTT Industrial Systems and VTT Processes. The particles in STYX bed follow the representative size distribution constructed for Olkiluoto case and the shape of particles is irregular. The bed depth is determined to be the same as expected in Olkiluoto plant if the whole corium inventory would spread uniformly on pedestal floor.

VTT Industrial Systems designed and built a test rig capable of heating a pressurised test bed immersed in water to a heat flux up to 1 MW/m^2 and run the tests. The resulting power and temperature distribution data is utilised by VTT Processes to model and assess the coolability concepts related to severe accident management in the Olkiluoto BWR.

The STYX-1 test-rig

The test vessel houses a particle bed having a diameter of 0.3 m and height of 0.6 m. The gravel bed is formed of a mixture of 0.2 to 10 mm particles, heated by an bare electrical resistance heating element in 6 levels 100 mm apart. The STYX test-rig is presented in detail in Figure 79.



Thermocouple and heating element levels (improved sensor set-up, 2nd test series).

T300/5 = level 300 mm from bottom, 5 elements
H050 = Heating element level 50 mm from bottom

1-4 feed-water system
5 pressure measurement
6 heating element
7 power measurement
8 pressure vessel
9-10 pressure control
11 water level indicator

12 condenser
13-17 condensate outlet
18 thermocouples (52)
19 flow meter
20 inner vessel
21 test bed

Figure 79. The STYX-1 dry-out heat flux test rig a) test bed and b) main components.

Heater element and power measurement

The heating element (nom. power 102 kW) is a cylindrically shaped three-phase resistance wire element (Kanthal D, Fe-Cr-AL alloy) distributed horizontally on six heating levels. The wire dimensions are 6 x 1 mm and the wire length per phase is 3.4 m. The element is supported by a steel rod frame and two support rings, held together by insulators. The heating element construction is shown in Figure 80. The heating element is powered by a thyristor unit and the power is regulated from the controller in manual mode, i.e. by setting the percentage of full output. The heater power configuration is fixed to generate a uniform power distribution to all six heater levels. The power distribution in the bed can not be changed in current test setup.

Temperature measurement

The thermocouples are distributed within the test bed as shown in Figure 79 in 14 levels adding up to 52 thermocouples within the test bed. Four thermocouples are located above the test bed immersed in the feed water for intended test with a small particle layer added on top. Additional thermocouples are located on the inner containment and pressure vessel structure.

Pressure and water level control

The test control variables, pressure, water level and mass flow are measured on-line. The pressure is followed from both a control gauge and an actual measuring gauge. The pressure control is performed with a control valve in the steam line followed by a plate heat exchanger with a capacity of 0.039 kg/s steam at 0.6 MPa. The condensed water is drained together with the condenser cooling water. The feed-water tank has a capacity designed for 10 h testing with full power. The water level is determined from a level gauge placed within a perforated container to minimise the turbulent behaviour of the boiling water on top of the bed. The water level measurement was used during the first tests for estimating the void fraction of the boiling water. The mass flow measurement showed to be indicative only due to mass flow meter sensitivity to the pressure control valve action.

Dryout heatflux test results

To this date two test series have been conducted. The first test series STYX-1 consisted of three tests, two at 0.3 MPa and one at 0.5 MPa pressure.

In all series-1 tests the first signals of temperature escalation and starting of dryout were received from the bottom-most thermocouples 4 cm below the lowest-level heater wire and 1 cm from the bottom of the particle bed. The formed dry zone was rather stable, only one thermocouple in a higher elevation, 10 cm from the bottom indicated expansion of dry zone in one out of three tests. In the 0.3 MPa tests first indications of dryout were observed at 35 and 26 kW power levels corresponding to dryout heat flux of 485 kW/m^2 and 383 kW/m^2. In the first test (35 kW) the local dryout was reached 8 minutes after a large power step of 12 kW, which may have exceeded the incipient dryout power. The second test gave dryout 3 minutes after a power increase of less than 3 kW, which is most likely closer to the incipient dryout power.

At 0.5 MPa pressure the dryout was encountered at power level 36 kW in 3 minutes following a power increase of 5 kW. The power was increased tree times after first indication of dryout in an attempt to investigate the expansivity of the dry zone. The dryout did spread to the next level at 47 kW (one sensor) but the test ended in heating element failure soon after final power increase to 50 kW. The heater temperature had exceeded its maximum operation temperature. Average void fraction in the bed was measured during the second 0.3 MPa and the 0.5 MPa test by turning off the power, and closing the feedwater and steam line valves and letting the water level collapse. The void fraction varied 0.2–0.28 with the measurement error of ±0.01. However, these measurements do not account for effects of stable steam pockets that may exist in the bed. Furthermore, the void fraction is not likely uniform in the bed, but the local void fractions could not be measured with the available equipment.

It is to be noted that the temperature sensor distribution in the first test series was less dense than described in Figure 79, i.e. evenly distributed between heating element levels (5 cm to element). The start of dryout was thus detected at a later point than with the denser sensor set-up in test series 2. New sensor levels were added during the forced service work required after the heating element failure, because the understanding and interpretation of STYX-1 results needed more accurate information about the location of the wet-dry interphase in the bed after dryout.

The second test series was conducted with the denser thermocouple set-up and a new heating element (as in Figure 79). A total of four tests were performed: at 0.5 and 0.7 MPa pressure, at near atmospheric pressure and last at 0.2 MPa pressure. In all the tests the dry-out was first detected 3–7 cm from the bottom of the test bed, 1–2 cm from the lowest heating element. The power and temperature histories of the atmospheric and 0.7 MPa tests are presented in Figures 80–81. After reaching the incipient dryout condition the temperature in the dry zone was highest near the heater wire, and the dry zone expanded both upwards and downwards, but not further up than to level 100 mm from the bottom during the 15–20 min holding time of the dryout power. The power was turned off before reaching a steady state configuration of the dry zone, beacause the temperatures near the bottom heater level were increasing continuously. The heat conductivity of alumina is low, 13.5–6 W/m/°C, in the temperature range of STYX-2 dry zone. The key results from second test series are gathered in Table 9.

Table 9. Key results of the STYX-2 dryout test series.

Pressure above atmospheric [bar]	Last power increase prior to dryout [kW]	Dryout power [kW]	Power hold-time before first indication of dryout in thermocouples [min]	Dryout heat flux [kW/m²]
0.15	1	16	15	232
2	1	21	25	300
4	6	26	10	371
6	1	32	7	451

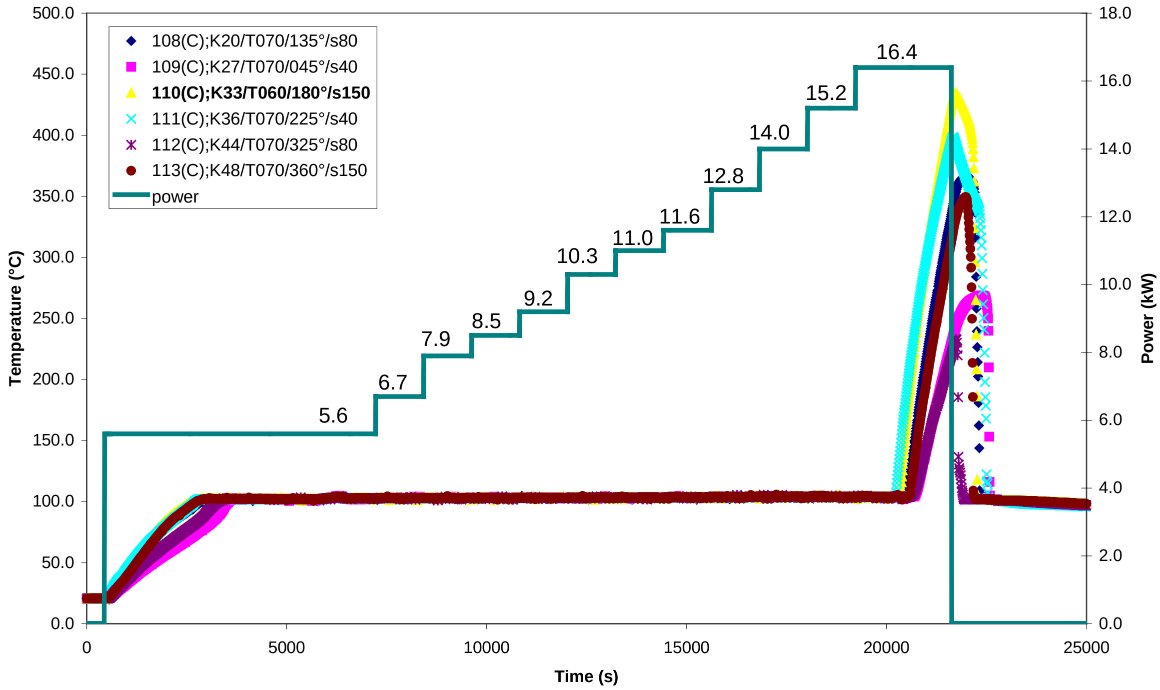


Figure 80. Test at atmospheric pressure (second test series, temperature history of level T70, 2 cm above the lowest heating element. The first thermocouple to detect the beginning of dryout was K36 at level 70 mm from the bottom of the bed.

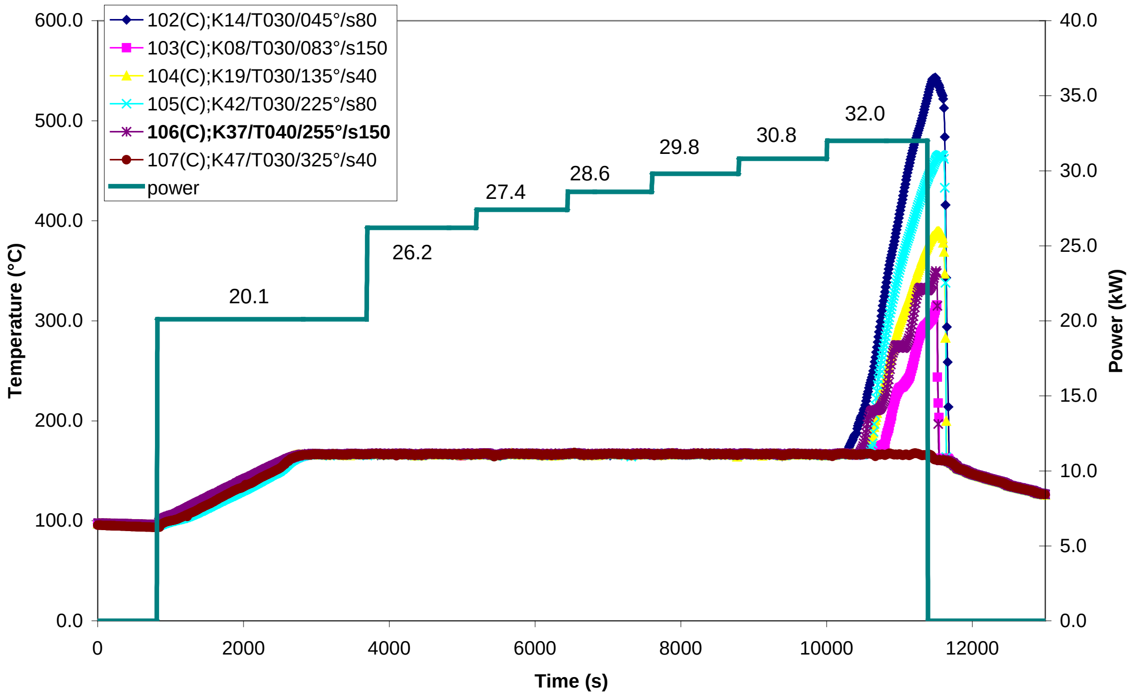


Figure 81. Dry-out test at 0.7 MPa (second test series), temperature history of level T30, 2 cm below the lowest heating element. The first thermocouple to detect the dryout was K37 at level 40 mm from the bottom of the bed.

The dryout heat fluxes measured with the STYX-1 facility were by a factor of 3 lower than predicted existing models for 3.5 mm particles. Also, the measured dryout heat fluxes from earlier experiments were generally higher, though the existing experimental data is rather scattered. In effort to understand the discrepancy between STYX-1 results and the model predictions, the effective particle size of the STYX-1 gravel was defined by measuring the permeability of the bed at the facility constructed at the laboratories of Fortum NS [7]. The permeability of the gravel was defined for air-simulating-steam and water -flow. The pressure loss across a 50 cm deep particle bed formed of STYX gravel was measured as a function of flow velocity. The measured values were fitted to the Ergun equation for particle beds (Eq. 15), with particle diameter being the unknown variable.

$$\frac{\Delta p}{H} = 150\frac{(1-\varepsilon)^2}{\varepsilon^3}\cdot\frac{\mu_f w}{(\phi d_p)^2} + 1.75\frac{(1-\varepsilon)}{\varepsilon^3}\cdot\frac{\rho_f \cdot w^2}{\phi d_p} \qquad (15)$$

where

Δp= measured pressure loss [Pa]
H= bed height [m]
ε= porosity of bed [–]

μ_f = dynamic viscosity of fluid [Ns/m^2]
w= measured superficial velocity [m/s]
ϕ= shape factor = 0.67 for irregular-shaped particles
d_p= particle size [m]
ρ_f= density of fluid [kg/m^3]

Both measurements yielded the effective particle diameter of 1.2 mm, which is much smaller than the mass-averaged particle size of 3.5 mm for the gravel. The application of 1.2 mm particle size in the models gives markedly better agreement with the measured dryout heat fluxes (Figure 82).

The configuration of a particle debris being spread uniformly over the whole floor area in the containment is the most conservative case for particle debris, but applicable design basis for debris coolability concept. If all core melt material in Olkiluoto plant is assumed to have spread on the pedestal floor following a pressure vessel failure, the estimated decay heat level would be about 0.73 % of the full power. The fraction of decay heat carried out from the melt with volatile fission products (noble gases, Cs and I_2) has been subtracted from the whole core decay heat. The pedestal floor area is 63.6 m^2, resulting in decay heat production per unit area of 287 kW/m^2. Figure 83 shows the comparison of measured dryout heat fluxes and the estimated heat generation per unit area for Olkiluoto plant. The STYX-1 and 2 series of experiments do not address the effects of a possible formation of a layer of finer gravel resulting from a more energetic fuel-coolant interaction on top of bulk debris. The effect of this so-called startified bed geometry will be investigated in the next phase, STYX-3, tests.

Conclusions

Dry-out tests have been successfully conducted at 0.1–0.7 MPa pressure for uniformly mixed beds with wide range of particles sizes and irregular particle shape. The STYX testing rig has been upgraded during the project to a point where a good estimation of the critical power resulting in a local dryout can be acquired. The dry zone formed in all tests near the bottom of the 60 cm-deep granular bed. The measured dryout heat fluxes increased with increasing pressure, obtaining a value of 232 kW/m^2 at near atmospheric pressure and the value of 451 kW/m^2 at 6 bar overpressure. When operating according to accident management procedures at Olkiluoto plant, the pressure in the containment plus the hydrostatic head from the pedestal water pool is 0.2 MPa. The dryout heat fluxes obtained with STYX facility were in general significantly lower than predicted by existing calculation models, if normally used mass-averaged particle size, 3.46 mm, was applied in the models. The experimental definition of effective particle size for irregular-shaped and deep STYX bed resulted in effective particle size of 0.804 mm, when the shape factor is taken into account. If the measured effective particle size is used in the dryout correlations, the calculated dryout heat fluxes are in good agreement with all STYX-2 measurements. The stratified bed geometry is likely to result in lower dryout heat fluxes, and it will be studied in the experiments currently under design.

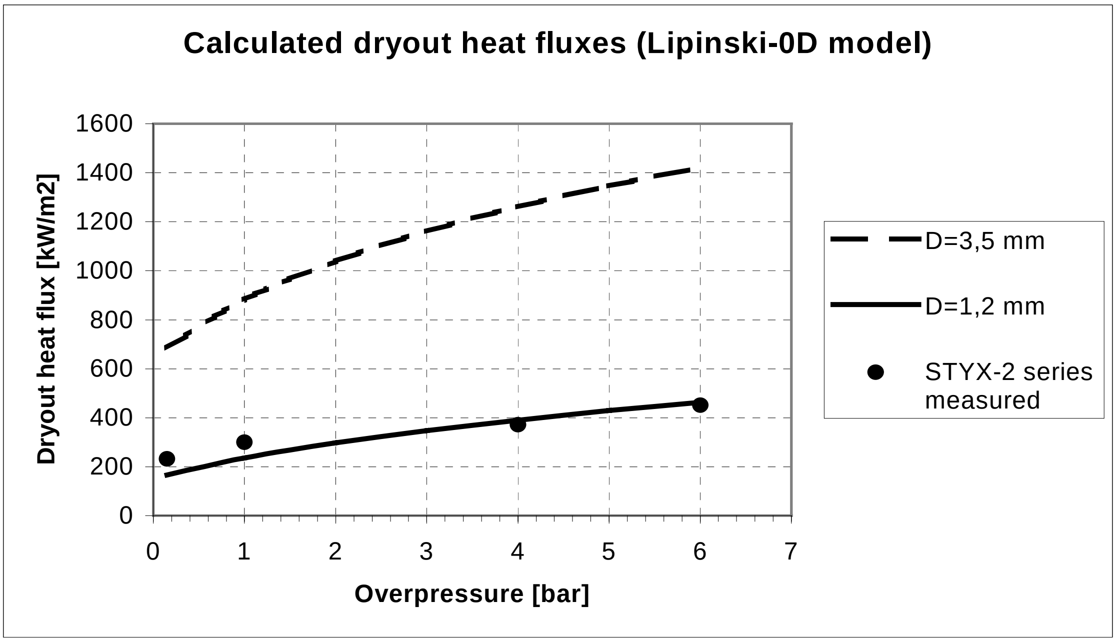


Figure 82. Lipinski-0D model prediction for STYX-2 dryout heat fluxes. Particle shape factor = 0.67 for irregular-shaped particles, bed porosity = 0.338.

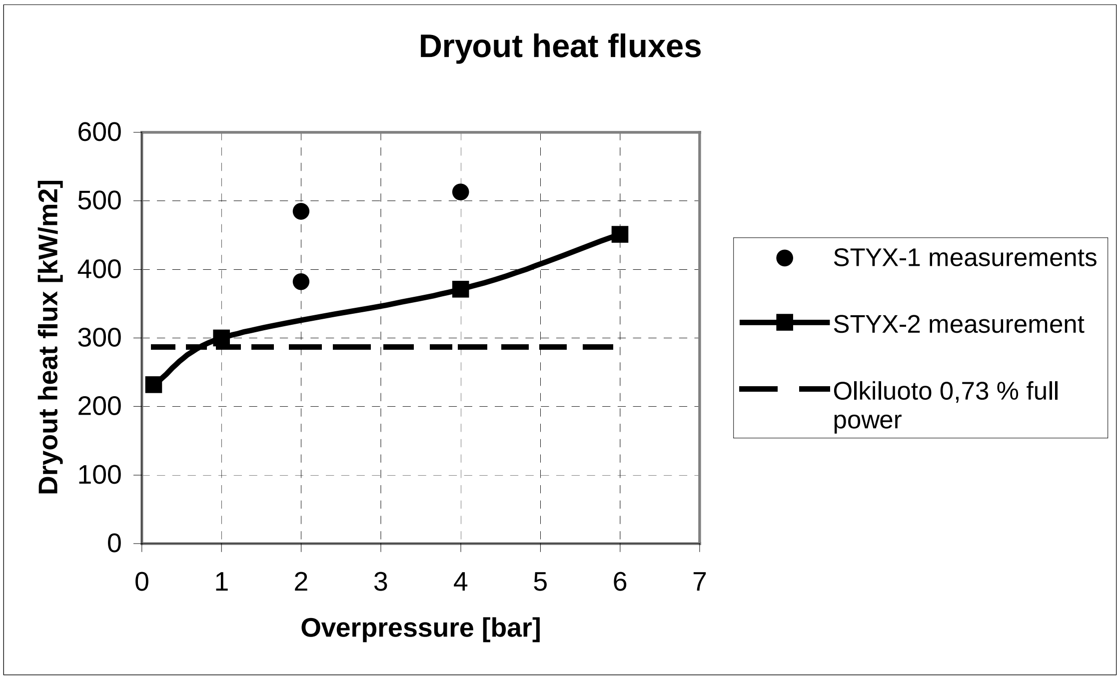


Figure 83. Comparison of dryout heat fluxes measured with STYX facility and estimated decay heat production in the particle debris bed of Olkiluoto plant.

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11. Fire safety research (FISRE)

11.1 FISRE summary report

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Abstract

Results of FISRE-project from years 1999–2002 are presented as a list and then selecting certain topics not published earlier. Critical heating times of metal case equipment were measured and modelled. A universal design curve was proposed for risk analysis. Exposing real and mock-up circuitry to smoke acute effects of smoke and humidity were studied on insulation resistance of electronics used in programmable automation. Relevant physical parameters of smoke and electrical performance of circuitry were measured online. A quantitative calculation model for smoke exposure, deposition on surfaces, and required protective layer performance is proposed. A new practical method for measuring of emissivity of metal surfaces was proposed, and tested by experiments. A Monte Carlo calculation platform PFS is presented for the distribution of the probability of the next target loss time in a cable tunnel in a NPP. Preliminary results using realistic distributions of the input parameters are shown. Use of database is widened for general fire scenarios. A new time lag model for smoke detectors is proposed. Modelling of active fire protection devices was started. Data collection and reduction work for extracting reliability data of fire detection and sprinkler systems in both nuclear and non-nuclear installations was carried out and preliminary analyses performed. Finally, applications of new performance based fire safety engineering are demonstrated by describing some recently constructed buildings.

Introduction

The level of fire risk analysis is still much behind of the level of other branches of PSA and needs further development. The main theme in the area was to improve the calculation tools for fire-PSA. 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. The project consists of several separate tasks, which all fit as small pieces into the greater puzzle.

Main objectives

The project was organised into three subprojects the titles of which cover roughly the fronts mentioned above: (1) effect of smoke and heat on electronics, (2) modelling of fire scenarios for PSA, and (3) active fire protection equipment.

The goal of (1) was to design and make experiments and physical models to explore acute effects of heat, smoke and humidity on parts of control equipment and electronics. The goal of (2) was to write a general calculation platform for determination of failure probabilities of selected important scenarios. This comprised setting up the compilation tools made from commercial pieces, using a general deterministic fire simulation model, implementing and developing submodels for certain components, and collecting statistical data of real situations from utilities. A subtask was to report on development of fire simulation advances worldwide. The goal of (3) was data mining and preliminary assessment of active fire protection devices from nuclear and non-nuclear installations. The first target was to obtain failure frequencies of components or subsystems as well as to propose specific models of the systems by adapting general reliability models according to data available. The major problem was to determine respective population sizes. For nuclear installations simple counting of populations were made, but for non-nuclear installations only fairly small samples could be obtained.

Main results

- For control equipment of NPPs contained within metal boxes a very simple general estimation formula of the critical heating time was obtained and validated for two instruments. In case of fire a sudden change of ambient temperature may occur. The critical temperature above normal ambient temperature is the maximum temperature, where the instrument still functions. The time when this temperature is reached is called the critical time span [1].

- Literature studies, experiments and modelling lead to fair quantitative understanding of acute effects of smoke, humidity or both on loss of insulation resistance of control electronics in NPPs. A quantitative calculation formula for the effectiveness of protective coating was proposed [2].

- A simple testing method for quantitative on-line assessment of smoke mass concentration was proposed [1].

- A new, simple and economic way of measuring emissivities at elevated temperatures from metal tube surfaces was proposed. Short series of experiments showed the method is useful, yields results at moderate accuracy and applies to cases, where thick metal samples are available [3].

- A Monte Carlo calculation platform PFS was made for estimating probability distribution for the time of damage of the second train in a cable tunnel of a NPP, given the ignition on the first train [4]. Realistic data were collected from utilities for both the cable tunnels and an electronic room, and both cases were calculated [5, 6]. The tool was released outside VTT for real use [6].

- By detailed modelling it was shown a finite ceiling jet velocity threshold is not a fluid dynamic effect [7]. A new theoretical closed form analytical model was proposed for the time delay of smoke detectors, which describes the physics of the underlying process [8].

- Reliability of fire detection system components was assessed for the first time using statistical data from nuclear [9] and non-nuclear [10] installations in Finland.

- Quantitative estimates on the reliability of sprinkler systems were extracted from statistical data on nuclear [11] and non-nuclear [12] installations in Finland.

- Large eddy simulation (LES) program FDS [13] was used to simulate a full-scale cable room scenario for international code comparison.

Longer descriptions of a few of the above results are shown below selecting those that were not presented earlier or are not included in the special article in this issue.

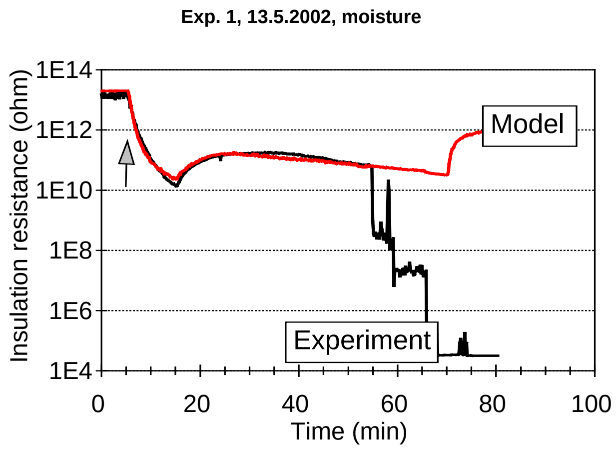
Insulation resistance of electronic circuits in smoke and high humidity

Acute smoke effects on electronics were reported earlier [14–16]. In a fire environment, high humidity can be created by any combination of the following: generation of moisture as a combustion product by the fire itself, high humidity in purge air used to remove smoke from the burning room and humidity from water suppression equipment used in fire-fighting. Moisture affects electrical circuitry in a negative way, insulation resistance between adjacent conductors is severely degraded by surface contamination in the presence of humidity. Humidity greatly increases the amount of damage that can result from smoke on electronics [14]. At high humidity, water combines with the salts (e.g. $ZnCl_2$) formed from the interaction between acid gases in the smoke and the metals, such as lead and zinc that can be part of the mechanical structure or solder in electronics. These salt solutions may be higher in conductivity than either the salt in a dry atmosphere or higher humidity atmosphere without smoke contamination [14]. Smoke works thus synergistically with humidity to increase leakage currents.

A distinct correspondence between the insulation resistance of the comb patterns and the relative humidity (RH) curves was noted in the experiments. An attempt was therefore made to model the influence of moisture on insulation resistance on comb patterns using a relationship proposed by Comizzoli [17] and assuming ohmic resistance between the conductors. The insulation resistance R is assumed to depend on relative humidity RH (%) and temperature T as

$$R = R_0 \exp(-A.RH)\exp(B/T) \tag{16}$$

where R_0 , A and B are here fitting parameters without physical interpretation. The only justification for choice of fitting parameters was to make the model curve fit by inspection. The model curve seems to fit quite well to the experimental curve (Figure 84).



$R_0 = 1 \times 10^{12}\ \Omega$

$A = 0.13$

$B = 1400$ K

Figure 84. Fit of model (equation (1)) for humidity influence on insulation resistance of comb patterns. Start of water vapour inlet is indicated with an arrow.

Both smoke alone and humidity alone caused a notable decrease in unprotected comb pattern insulation resistance, about 3 orders of magnitude. The combined effect of moisture and smoke was still larger, about 5 decades decrease in insulation resistance.

Emissivity of metal surfaces

Fire safety in nuclear power plants depends, among other things, on the capability of the plant instrumentation and control system to perform its duties during the fire. One of the factors to be considered is the heat transfer from the fire to stainless steel piping which forms a part of the instrumentation system.

There are no simple means of determining emissivity for the total thermal radiation range. A calorimetric method was designed using known principles [18] and emissivity of a stainless steel tube was studied experimentally using an electrically heated laboratory furnace. The test equipment worked well, and was capable of producing good data with reasonable effort [19] as shown in Figure 85.

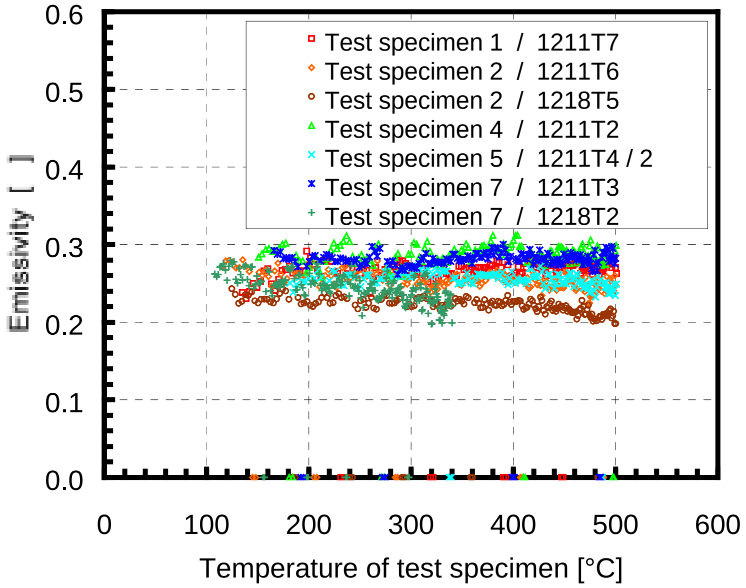


Figure 85. The emissivity of stainless steel tube surface as a function of tube temperature. Data obtained using veteran test specimens (test specimens that have already been exposed many times to a high temperature).

Smoke detector time lag

To estimate the flow of smoke through a detector the problem is divided into two formally different modes: flow of air, and flow of smoke particles [8]. From airflow the continuity equation in steady state form yields the pressure inside the detector. Once it is known, filtration theory [20] can be applied in the sense of perturbation theory to calculate penetration of smoke particles through the outer protective screen to estimate the detection time.

Through lengthy calculus a formula follows in closed form for the penetration of smoke particles [8]. Figure 86 depicts relative penetration as a function of ceiling jet velocity for monodisperse smoke of given particle size. It shows that at small velocities penetration becomes practically zero explaining the finite critical velocity observed. This theoretical proposal suggests smoke is not a simple fluid for such a transport calculation, but a real aerosol, where permanent fluid (air) flows differently as solid suspended smoke particles. Although the proposed model is based on a wide series of good experimental work, it is still a theoretical construction. Direct experimental measurements in conditions close enough to real situation of smoke detection are needed to test the validity of the model.

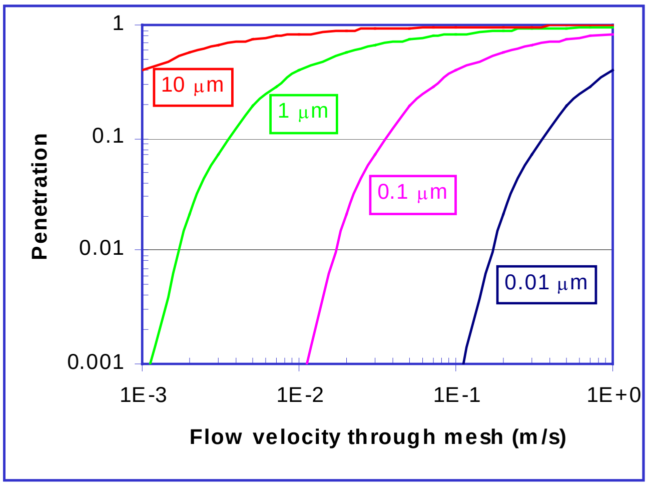


Figure 86. Particle penetration through a mesh as a function of flow velocity for different diameters of smoke particles.

Applications

Since this research has been strategic and going deep into new information, direct applications of are not very dominating in the results of our project. Although most of the methods developed are ready for application, the targets are mainly inside NPPs and not well known to public. Monte Carlo program FDS was given to utilities for real use [6]. Sprinkler data have also been applied to real construction cases [21].

As a cue to everyday problems visible to great public is a recent paper [22]. There through some newly constructed well known buildings is demonstrated, where new performance based fire safety engineering is used. Much of the same advanced techniques can be used on nuclear and non-nuclear installations. Although historically, nuclear energy related research has been a spearhead also in fire research this is not any more generally true. There are several areas, where even strictly nuclear approaches have to be supplemented by non-nuclear elements. It has taken place in FISRE project in the theme of reliability of active fire protection equipment.

Conclusions

The strategic goal of this project has been to improve quantitative capability of fire-PSA in NPPs. Using technology transfer, experiments and modelling of phenomena new deeper understanding of fire problems has been obtained. Furthermore, new measuring and smoke detection techniques have been proposed. The methods developed are not nuclear exclusively, but find wide application also outside nuclear field as shown by examples of recently constructed buildings.

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11.2 Probabilistic fire simulator – Monte Carlo simulation tool for fire scenarios

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Abstract

Risk analysis tool is developed for computing of the distributions of fire model output variables. The tool, called Probabilistic Fire Simulator, combines Monte Carlo simulation and CFAST two-zone fire model. In this work, it is used to calculate failure probability of redundant cables and fire detector activation times in a cable tunnel fire. Sensitivity of the output variables to the input variables is calculated in terms of the rank order correlations.

Introduction

Traditionally, the deterministic fire models have been used to estimate, what are the typical consequences of the fire, at the given values for the input parameters. The goal of this study is to develop a tool for the probabilistic fire simulation. A risk analysis model is developed using the well-known technique of Monte Carlo simulations. A two-zone model CFAST [1] is used to model smoke spreading and gas temperature during the fire. The risk analysis model and the physical models are combined in a spreadsheet computing environment. The tool, called as Probabilistic Fire Simulator (PFS), is intended to be fully general and applicable to any fire scenario amenable to deterministic numerical simulation. The distributions of the selected result variables, for example component failure time, are automatically generated. The sensitivity of the output variables to the input variables can be calculated in terms of the rank order correlations. The tool is used to model a fire in a nuclear power plant cable tunnel, which was previously studied experimentally by Mangs & Keski-Rahkonen [2]. Here, the effects of the tunnel geometry and fire source properties are studied by fire modelling and variation of the input variables. The input distributions are based on the statistics collected from the power plant. Other applications, such as office fire and electronics room fire, were studied during the project as well.

Monte Carlo simulation

The question set by the probabilistic safety assessment process is usually: “What is the probability that a certain component or system is lost during a fire?” This probability, denoted here by P_{loss}, is a function of all possible factors that may affect the development of the fire and the systems reaction to it. Similar systems on the other fields than fire have been studied extensively [3]. Here we adapt this general theory for our specific fire problem.

Let us denote the group of affecting variables by vector $\mathbf{X} = \left(X_1 \;\; X_2 \;\; ... \;\; X_n \right)^T$ and the corresponding density functions f_i and distribution functions F_i. The loss of the target component/system is indicated by a limit state function $g(t,\mathbf{x})$, A limit state condition is

$$\begin{aligned} &g(t,\mathbf{x}) \le 0, \text{if component/system is lost at time } t \\ &g(t,\mathbf{x}) > 0, \text{if component/system is not lost at time } t \end{aligned} \tag{17}$$

The probability of loss can now be calculated by integral

$$P_{loss}(t) = \iint\limits_{\{\mathbf{x}|g(t,\mathbf{x})\le 0\}} \cdots \int \phi_x(\mathbf{x}) dx_i \tag{18}$$

where ϕ_X is the joint density function of variables $\mathbf{X}$. In this work, P_{loss} is calculated using Monte Carlo simulations where input variables are sampled randomly from the distributions F_i. If $g(t,\mathbf{x})$ is expensive to evaluate, stratified sampling technique should be used. In Latin Hypercube sampling (LHS) the n-dimensional parameter space is divided into N^n cells [4]. The advantage of this approach is that the random samples are generated from all the ranges of possible values, thus giving insight into the tails of the probability distributions. A procedure for obtaining a Latin Hypercube sample for multiple, spatially correlated variables is given by Stein [5]. He showed that LHS will decrease the variance of the resulting integral relative to the simple random sampling whenever the sample size N is larger than the number of variables n.

The sensitivity of the output to the different input variables is studied by calculating the Spearman’s rank-order correlation coefficients. A value’s “rank” is determined by its position within the range of possible values for the variable. RCC is then calculated as

$$RCC = 1 - \frac{6\sum d^2}{n(n^2-1)} \tag{19}$$

where d is the difference between ranks of corresponding data pairs, and n is the number of data pairs. RCC is independent of the distribution of the initial variable.

Fire modelling

The transport of heat and smoke is simulated using a multi-room two-zone model CFAST [1]. It assumes two uniform layers, hot and cold, in each room of the building and solves the heat and mass balance equations for each room. PFS worksheet is used to generate the input data for CFAST. The user may combine any experimental information or functions to the fire model input. The most important source term in the simulation is the rate of heat release (RHR). The RHR can be defined using analytical curves, such as t^2-curve [6], or specific experimental curves.

Typical results of the fire simulation are gas temperatures, smoke layer position and temperature of some solid object such as a cable. Usually, the actual target function is time when some event takes place. Examples of the target functions are smoke filling time, flash over time and failure time of the solid objet. In this work, the most important target function is the cable failure time. An analytical, time dependent solution of the axially symmetric heat transfer equation is used to calculate the cable core and surface temperatures.

Results and discussion

A fire in the cable tunnel of a nuclear power plant is simulated to find out the failure probability of cables located in the same tunnel with the fire. An effect of possible screen that divides the tunnel between the source and target is studied. A plan view and vertical cut of the physical geometry and the corresponding CFAST model are outlined in Figure 87. The tunnel is divided in five virtual rooms to allow horizontal variations in layer properties. The fire source is located in ROOM 1 and the target cable in ROOM 3, just on the opposite side of the screen. The fire source is modelled using an analytical t^2-type curve with a randomly chosen growth time t_g and maximum level $\dot{Q}_{\max}$, that depends on the tunnel size.

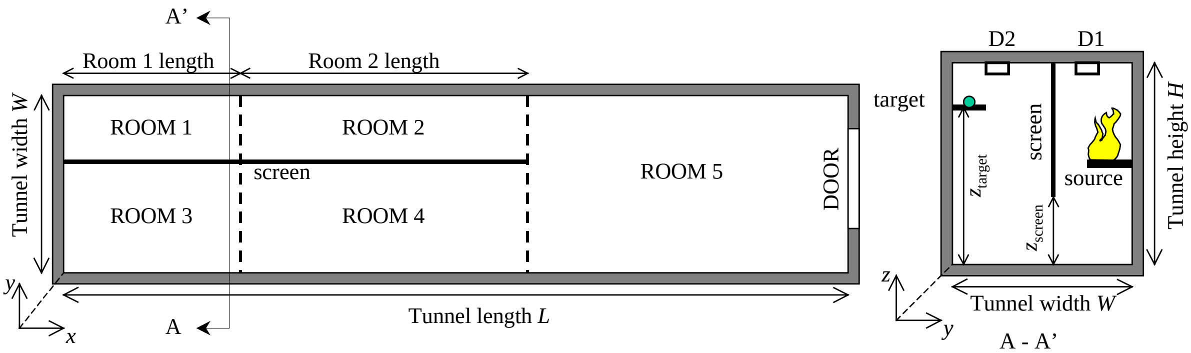

Figure 87. A plan view and vertical cut A-A' of the cable tunnel model. D1 and D2 denote heat detectors, located in ROOM1 and ROOM3, respectively.

The dimensions of the tunnel and the cable tray locations are taken from the measured distributions of the power plant. About 50 tunnel cross sections have been studied to generate the distributions. The distribution of the cable diameter is based on the measurements in seven tunnel cross sections, containing 815 cables. A complete list of the random variables is given in Table 10. While most of the variables are true physical properties and dimensions, the lengths of the virtual rooms are purely associated to the numerical model. Three output variables are here considered: Failure time of the target cable and activation times of the heat detectors D1 and D2.

The Monte Carlo simulations were performed to generate the distributions of the output variables, and to find out the importance of each input variable. The convergence of the simulation was ensured by monitoring the values of the 10, 20, ... 90 % fractiles, mean values and standard deviations of the output variable distributions. About 1000 iterations were needed to reach the convergence. The simulations took typically about one day on a 1.7 GHz Pentium Xeon processor.

Table 10. A list of random variables.

Variable	Distribution	Mean	Std.dev	Min	Max	Units
RHR growth time t_g	Normal	1000	300	0	3000	s
Source height z_{source}/H	Uniform			0	0.7	
Ambient temperature	Normal	20	3.0			°C
Tunnel height H	Measured					m
Tunnel width W_{tunnel}	Measured					m
Room 1 length	Uniform			2	5	m
Room 2 length	Uniform			5	10	m
Tunnel length L	Uniform			30	100	m
Door height z_{door}/H	Uniform			0.1	1.0	
Door width W_{door}/W_{tunnel}	Uniform			0.01	1.0	
Screen edge height z_{screen}/H	Uniform			0 / 0.95	0.5 / 1.0	
Dimensionless cable height z_{cable}/H	Measured					
Cable radius	Measured					mm
Critical cable temperature	Normal	200	20			°C
Cable conductivity	Normal	0.16	0.05	0.1	0.5	W/Km
Cable density	Normal	1400	200	1000	2000	kg/m^3
Detector activation temperature	Normal	57	3			°C
Detector RTI	Normal	50	10	40	60	(m/s)$^{1/2}$
Ventilation time constant	Uniform			0.5	10	h
Concrete density	Uniform			1500	3000	kg/m^3

The distributions of the target failure times are shown in Figure 88. The overall failure probabilities can not be derived from the simulations because the distributions do not reach their final values during the total simulation time of 2400 s. However, the existence of the screen is found to dramatically decrease the failure probability of the redundant cable. The distributions of the activation times of fire detector D2 are shown in Figure 89. The screen decreases the activation probability from 1.0 to 0.8. In Figure 90 the detector D1 activation times are plotted against the failure time, demonstrating that the fire is always detected before the failure.

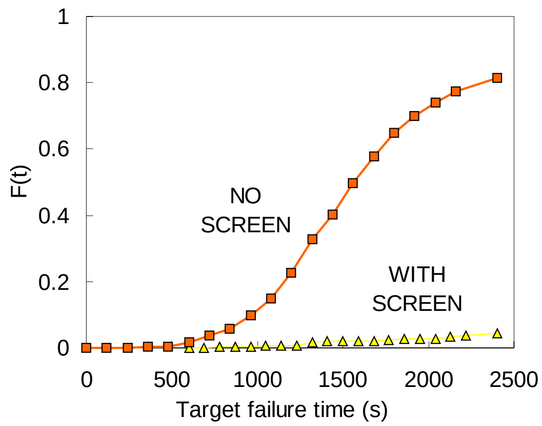

Figure 88. Target failure time distributions.

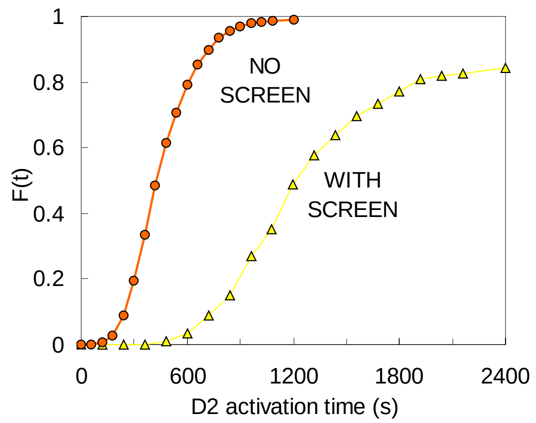

Figure 89. D2 activation time distributions.

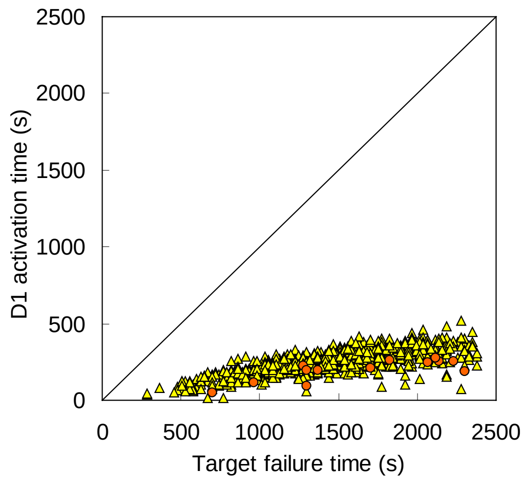

Figure 90. D1 activation times vs. target failure times (N = 954).

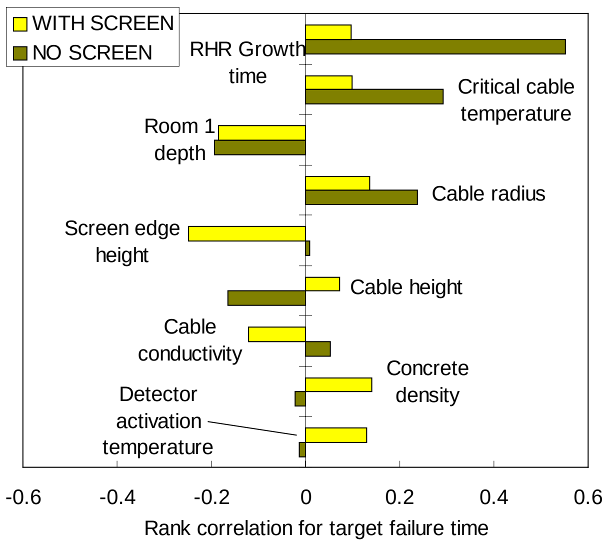

Figure 91. Rank correlations for target failure time.

The sensitivity of the target failure time for the various input variables is studied by calculating the rank order correlations. Nine most important coefficients are shown in Figure 91. RHR growth time and the critical temperature of the target cable are important, if screen is not present. The length of the virtual fire room has a strong correlation with the failure time in both cases. Various ways to decrease the importance of this numerical parameter should be studied to improve the reliability of the results. In the scenario where screen is present, the height of the lower edge of the screen has strong negative correlation with the failure time. It is therefore recommended, that the screen, when present, should reach as low as possible to prevent smoke from flowing under the screen. The rest of the correlations are not significant.

Conclusions

Probabilistic Fire Simulator is a tool for Monte Carlo simulations of fire scenarios. The tool is implemented as a worksheet computing tool, using commercial @RISK package for Monte Carlo part. The fire is modelled using two-zone model CFAST. The Monte Carlo simulations can provide the distributions of the output variables and their sensitivities to the input variables. Typical outputs are, for instance, the times of target failure, fire detection and flashover. The tool can also be used as a worksheet interface to CFAST. Extension to the other fire models is also possible.

The presented example case demonstrates the use of the tool. The results of the tunnel scenario show that the heat detector gives an alarm before the loss of the redundant cables, with a very high probability. However, the detector reliability was not considered. According to the sensitivity measures, the most important parameters for the safety of the redundant cables are *i*) the growth rate of the fire, *ii*) the screen providing a physical separation of the burning and target cable trays, *iii*) the critical temperature of the cable material and *iv*) the radius (mass) of the cable. Unfortunately, the failure time is also sensitive to the length of the virtual room of fire origin, which is a purely numerical parameter. This phenomenon should be studied more carefully in the future.

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12. Programmable automation system safety assessment (PASSI)

12.1 PASSI summary report

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Abstract

In the licensing of a programmable automation system different kind of evidence is collected and analysed in order to make sure that the system fulfils the safety requirement set for the system by the authority. The PASSI project aims to provide support for the authorities and utilities in the licensing process of programmable automation systems. This report presents a summary of the tasks and results gathered in the PASSI project.

Introduction

In the field of nuclear industry the technical and economical ageing of analogue automation systems is causing more and more pressure for their replacement with corresponding software-based automation systems. The technological leap in a rather well-established industry might for the beginning sound simple but is indeed everything but simple. The extra trouble in the transfer from analogue to software-based automation systems has two main reasons. First, strict reliability goals are set for automation systems responsible for safety-critical functions in nuclear power plants and, to achieve these goals, the systems should be well built and thoroughly tested. Second, the discontinuous behaviour of discrete logic in software-based systems has the effect that to be sure on the functionality of the system in all occasions the system should be tested with all possible inputs. However, this is usually not feasible because of the large number of possible inputs even for a simple system and, therefore, for a more complicated software-based system the thorough testing would require unacceptable amount of time and effort. One proposal given to overcome the difficulty is to compensate the shortage from testing with reliability related evidence from other sources closely involved with software-based systems.

Main objectives

In general the PASSI project aims to provide support for the authorities and utilities in the licensing process of programmable automation systems. The objective of the project is carried out by acquiring, developing and testing new and more cost-effective methods, tools and practices for the safety and reliability assessment of software-based automation systems. The practical experience gathered from the use of methods, tools and practices in the project helps to achieve a more streamlined licensing process for programmable automation systems. Part of the project dealt with the ageing phenomenon of modern software-based automation systems.

Reliability assessment methods

The work in this area has mainly focused on the development of reliability assessment methods and practices for software-based systems. The emphasis has been on the quantitative reliability modelling leaving, however, room for qualitative characterisations as well. Tools related to the methods and practices were tested on **an experimental application study of estimating the reliability of a software-based motor protection relay**. The study is presented in more detail in the special report below. One of the purposes in the experimental application study was the recognition of further development needs and at the creation of a proper base for the justification of safety claims in the licensing process of programmable automation systems. For closer review on the topic see Helminen and Pulkkinen [1].

The reliability assessment of software based systems (e.g. for the plant PSA analyses) is a difficult task and no widely accepted, cost-effective methods so far exist. With the reliability of software-based systems there are certain characteristics that makes it difficult to utilise the same reliability assessment methods used for traditional analogue automation systems. New assessment methods are needed and one important feature in the assessment methods for software-based systems is that they enable the combination of disparate reliability evidence consistently and flexibly together.

While achieving high reliability for a system is important, at least as important is how to assess this reliability. How to explain to other people the reliability of the system? How to give justified claims about the reliability that are consistent and traceable? A valid assessment method should enable the solid reasoning and elicitation of the reliability claims and give the maximum support for the decision making in the licensing of programmable automation systems.

The main sources of reliability evidence for software-based systems are depicted in Figure 92. Part of the evidence may be directly measurable statistical evidence e.g. an evidence obtained from operational experience and testing. Part of the evidence may be of qualitative nature as it often is in the case of design features and the development process of a system. In the experimental application study evidence from the system development process and the operational experience of the system was used for the reliability estimation. The estimation was based on **Bayesian statistics and in**

particular to its technical solution called Bayesian networks. Bayesian statistics enable the implementation of qualitative and quantitative information flexibly together as well as updating the estimation while new information is obtained about the system and introduced to the estimation. Bayesian networks enable also consistent and transparent reasoning about assessor's belief on the relation of different pieces of evidence. For more detailed review on the topic see Helminen [2].

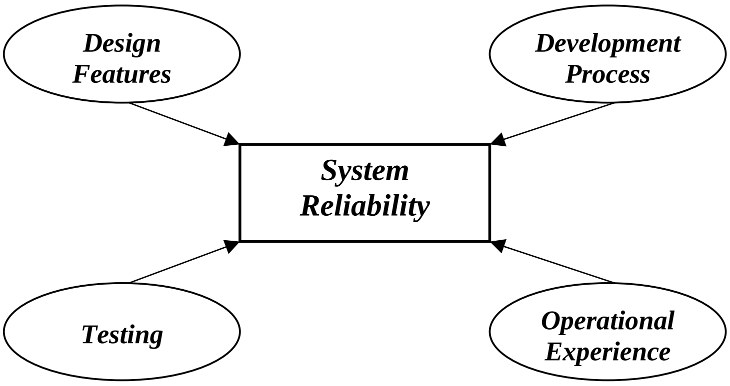

Figure 92. Main sources of reliability evidence for software-based systems.

Based on the evidence sources used in the experimental application study it could be described as an operational experience analysis even though other elements of evidence are involved in the study as well. Another analysis under study in the project has been **the failure mode and effects analysis (FMEA) of software-based systems**. The failure mode and effects analysis is one of the well-known analysis methods having an established position in the traditional reliability analysis. The purpose of FMEA is to identify possible failure modes of the system components, evaluate their influences on system behaviour and propose proper countermeasures to suppress these effects. In the project a literature study on FMEA and a study on the use of the method for the reliability analysis of software-based systems was carried out. As a result of the study it was concluded that the FMEA of software-based systems is applicable only to a limited extent providing, however, an important addition to the reliability assessment of software-based systems. FMEA is especially advantageous in revealing the possible weak points of the systems, and can therefore help with the generation of the tests cases for the system. For the full report on the subject see Haapanen and Helminen [3].

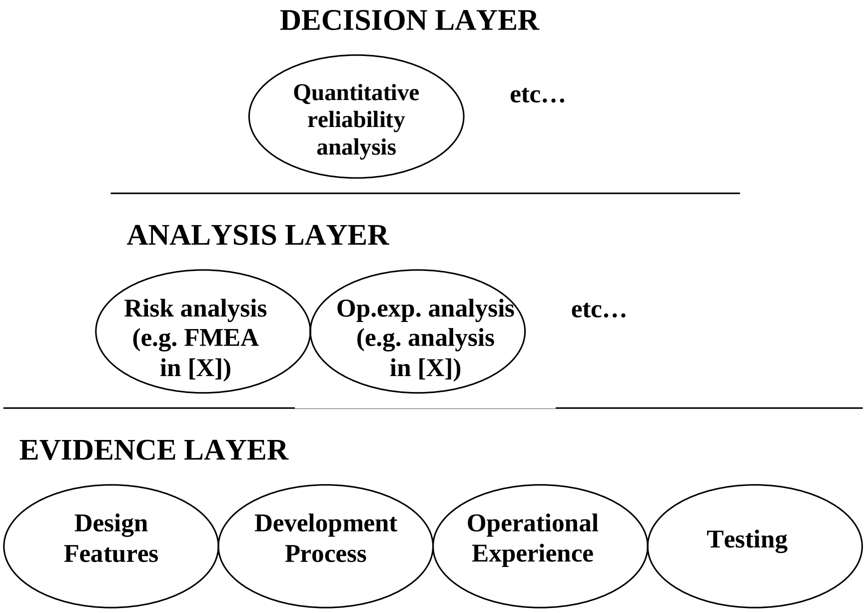

Figure 93. Reliability assessment pyramid (uncompleted).

The analysis studied and developed in the project form the foundation for refining the information obtained from the evidence of software-based systems. The best result is obtained if the methods used in the analysis can be easily connected to the decision making in the licensing process of programmable automation systems. The situation can be seen as a pyramid of three layers as illustrated in Figure 93. At the bottom is the evidence layer embodying the evidence sources of Figure 92. Above the evidence layer is the analysis layer containing a collection of different analyses relevant for the reliability assessment of a software-based system. Top of the pyramid is the decision layer, which gathers the information provided and cultivated by the different analyses of the analysis layer. The advantage of **the reliability assessment pyramid** of Figure 93 is that the process of reliability assessment for a software-based system, or, for a part of the system, can be carried out using a rational and consistent structure where all the important features of reliability are covered, leaving at the same time an easy access to the source evidence. More detailed discussion of the research topic will be given in a report to be completed by the end of year.

Ageing related failure modes of modern I&C equipment (ICAGE)

The work in the area started with a pre-study where a literature survey on the earlier analyses of instrumentation and control (I&C) equipment were reviewed and the needs of power utilities and authority on the subject was identified. Based on the pre-study more detailed project plan was established.

One of the main purposes of the ICAGE project has been in the gathering of information on the commonly used modern I&C equipment. Information on the components and their materials is collected and technologies, requirements and standards related to the modern I&C equipment are defined. Special focus has been in the influence of the new environmental directives of EU (WEEE, EEE and RoHS) and in the identification of valid acceptance and reliability testing methods for the ageing of modern I&C equipment. The information is partly gathered by participating in an international conference on the management of ageing of nuclear power plant instrumentation and control equipment.

Based on the pre-study and the information gathered during the ICAGE project a preliminary plan of reliability tests for 2 or 3 different operational conditions is designed. In the test plan the necessary initial data of the electronic components and the materials are evaluated. Also the amount of testing time for the simulation of certain operational ageing is estimated.

Benchmark exercise on safety evaluation of computer-based systems (BE-SECBS)

The BE-SECBS project belongs to the 5th Framework Programme of the European Atomic Energy Community. It is not directly performed as a part of the PASSI project. However, the steering group of the FINNUS program decided that BE-SECBS is discussed and followed up by the PASSI project. The work carried out in BE-SECBS can be seen as an application of the results of the PASSI project.

The BE-SECBS project started in the beginning of year 2001. During the first two years of the project, the benchmark exercise has been specified and the system under case study in the project has been designed and realised. The task of VTT and STUK is to assess the safety of the case study system using a method preliminarily described for the purpose. The method provides both qualitative and quantitative reliability evaluation results of the case study system. The steps in the STUK-VTT methodology include the analysis of requirement specification, analysis of documentation provided by the system designer, additional analyses (e.g. failure mode and effects analysis and test coverage analysis), analysis of operating experience and additional testing of the system. The quantitative reliability evaluation of the system is strongly connected to the Bayesian network models applied in the reliability assessment methods subproject. The BE-SECBS project will conclude at the end of June 2003 (according the revised project plan).

Along with the benchmark exercise another international connection in the PASSI project has been the co-operation with the OECD Halden Reactor Project. Closer introduction to the reliability assessment co-operation is given in two reports by Gran & Helminen [4][5].

Applications

The automation renewal in Finnish nuclear power plants is underway. In fact, the approval of new automation systems for Loviisa power plants will occur in the next few years. Similar approvals of new automation systems on a smaller scale have already been made at Olkiluoto power plants. The major automation renewal of Olkiluoto power plants will follow a couple years after Loviisa. At this point it is clear that the new automation systems will be based on programmable technology. This will also be the case with the automation of the fifth power plant to be built in Finland. Renewal of automation systems in the nuclear power plants will probably be the biggest individual change since the building of the plants.

A practice embodying methods that could be performed cost-effectively and would guarantee that the system subject to the licensing will reliably enough execute the safety functions addressed to it are still under discussion. Experience from other nuclear power plants provide only little support for the problem since, even in worldwide scale, there is only little experience on the licensing of programmable automation systems. STUK has published a guideline on the licensing of programmable automation [6]. The guideline gives a framework for the licensing process of programmable automation and the PASSI project gives support for the practical implementation of this framework.

Conclusions

In the PASSI project new reliability assessment methods and practices for software-based systems have been developed and tested. With the new assessment methods the special needs of the reliability assessment of software-based systems are taken into consideration. The key assessment method developed and tested in the project is based on Bayesian statistics and in particular to its technical solution called Bayesian networks. Bayesian networks enable the implementation of reliability evidence from disparate sources flexibly together same time providing consistent way of reasoning assessor’s beliefs on the relationship of the different kinds of evidences. The assessment method makes possible the solid reasoning and elicitation of the reliability claims and that way gives maximum support for the decision making in the licensing of programmable automation systems.

The assessment method developed in the project was tested in an experimental application study on a qualitative reliability estimation of a software-based motor protection relay. The results on the use of the method were encouraging and further testing of the methodology will be practised in the benchmark exercise next year. A study on the failure mode and effects analysis of software-based systems was carried out and the results of the study will be utilised in the future example cases. Information related to the ageing of modern I&C equipment was gathered and a preliminary test plan for the simulation of certain operational ageing was prepared.

There is an immediate need of the assessment methods and practices for software-based systems in the Finnish nuclear power plants with an urgent time schedule of the automation renewal.

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12.2 Reliability assessment of a software-based motor protection relay using Bayesian networks

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Abstract

Often to make justified reliability claim of a certain system different kinds of evidence needs to be combined. Some of the evidence supporting the claim may be of qualitative type, whereas some of the evidence may be of quantitative type. Combination of disparate evidence together is not always straightforward and the reasoning behind the conclusions obtained from the combination may be hard to explain. Bayesian networks provide a consistent and transparent method for the combination of the evidence and for the reasoning of one's beliefs on the relation of different pieces of evidence. In the special report we demonstrate the combination of disparate evidence with a case study on the reliability assessment of a software-based motor protection relay, where the combination of the reliability related evidence has been carried out using Bayesian networks. The reliability related evidence in the case study is the expert judgement on the development process and the operational experience estimated for the software-based motor protection relay.

Introduction

The need for shifting from analog instrumentation and control (I&C) systems to corresponding digital I&C systems is becoming current also with the I&C systems of nuclear power plants. To have a control over this transition and to have basis for the licensing of digital I&C systems there needs to be solid methods for assessing the reliability of the new digital I&C systems. However, to give the maximum support for the licensing process of digital I&C systems the reliability assessment method should not only produce exact values related to reliability, but with the assessment method the reasons for the produced values should be clarified.

With the analog systems there is the nice feature that systems can be tested thoroughly. For digital systems this is usually not the case. Digital systems mainly fail because of their inherent design faults, which are triggered at appropriate conditions with certain inputs. Since the number of possible inputs for even relatively simple systems becomes unreasonably large, the system can only be tested to a certain extent. As the system is not tested with all the possible inputs under all different conditions i.e. under all operational profiles, there is always some uncertainty left to the reliability of the system.

The most convenient way to handle such uncertainty is through the use of probabilistic calculus.

For digital systems that can't be tested exhaustively there rises a question if such systems can ever be used in applications for which high reliability is required. One way to tackle this dilemma is to compensate the lack of reliability related testing evidence with evidence from other sources. Such sources are for example the system development process, design features and operational experience. In the reliability assessment of digital systems this compensation of evidence is not always a straightforward operation. Different evidence sources involve many qualitative characteristics that are difficult to translate to unambiguous quantitative measures. These characteristics, however, contain a lot of information relevant to the reliability of the system and should not be overlooked and thrown away. Therefore, the methods used for the reliability estimation should have enough flexibility to overcome the problem.

The approach taken in the report is based on Bayesian statistics and in particular to its technical solution called Bayesian networks. Bayesian statistics enables the implementation of qualitative and quantitative information flexibly together as well as updating the estimation while new information is obtained about the system and introduced to the estimation. As well as combining evidence Bayesian networks also enable transparent reasoning about one's belief on the relation of different pieces of evidence. The reliability assessment presented in this special report is a summary of a report by Helminen and Pulkkinen [1]. The theory and ideas on which the assessment is mainly based on can be found more explicitly explained in report by Helminen [2].

Combination of expert judgement and operational experience

In the reliability assessment the qualitative and quantitative reliability related evidence of software-based system has been combined using Bayesian networks. The combination follows a pattern where the qualitative evidence has been used in the building of the prior estimation for the reliability of the system while the quantitative evidence is used to update the prior estimation. In the case study the qualitative evidence is based on the expert judgements of the system developers on the development process of the system. The qualitative evidence is quantified in an expert judgement process described below. The quantitative evidence in the case study is the approximated operational experience of the system during its life cycle.

A diagram representing the building of a prior estimation for the example case using expert judgement is presented in Figure 94. The expert judgement process is divided to six steps. In the first two steps the experts give score values and weights for the different software development phases of the system. In the third step total score values for each expert are obtained using an additive value function. To convert the total score value of the expert to a corresponding failure frequency distribution, in step four, the experts are asked to give failure frequency distributions for two or three different score values. Based on the total score value determined in step three and the failure frequency distributions of different score values given in step four, in step five a prior failure frequency distribution of the system is determined for each expert by using the method of least square. In the last step the different failure frequency distributions given by the experts are combined to one joint prior distribution reflecting the reliability of the software of the target system.

The operational experience in the example case has been the approximated amount of working years and the amount and types of software faults encountered with the different software versions of the system. The combination of evidence from expert judgement and operational experience is carried out by the basic principle of Bayesian network theory. The combined prior distribution is modelled as a mixture of lognormal distributions. The software faults, on the other hand, are found in certain time interval and therefore follow a Poisson distribution. Therefore, the underlying model in the Bayesian network is so called lognormal-Poisson model. Closer review on using lognormal-Poisson model in failure frequency rate estimation can be found for example in report by Pulkkinen and Simola [3].

Case study on the reliability assessment of a software-based motor protection relay

Target system and assessment process

The system under assessment in the case study is SPAM 150 C -motor protection relay produced by ABB Substation Automation. The numerical motor protection relay SPAM 150 C is an integrated design current measuring multifunction relay for the complete protection of alternating current motors. The main area of application covers large and medium-sized three-phase motors in all types of conventional contactor or circuit-breaker controlled motor drives.

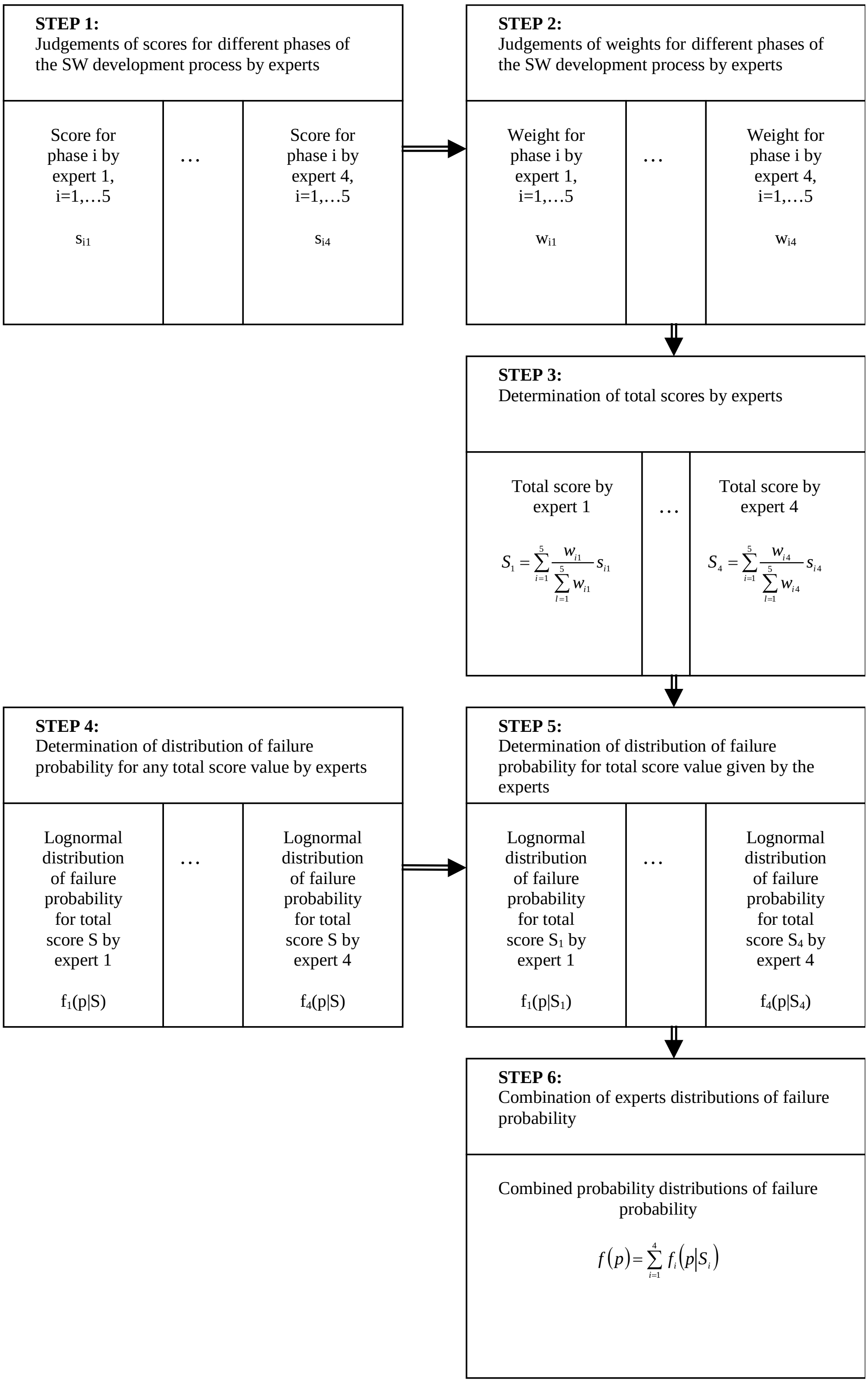

Figure 94. Diagram presenting the six steps of the expert judgement process.

In the assessment process the whole life cycle of the motor protection relay is taken into consideration. A diagram representing the assessment process is depicted in Figure 95. First, a prior estimate for the reliability of system's first software version is built. The prior estimate is based on the expert judgement process explained above. The estimate is then updated using the data, i.e. operational experience, obtained for the first software version. Later on, when the software is modified, the effects of the modifications to the system reliability are estimated and operational experience for the second version is included in the estimation. This procedure is repeated, as many times as there are new software versions produced in the life cycle of the system.

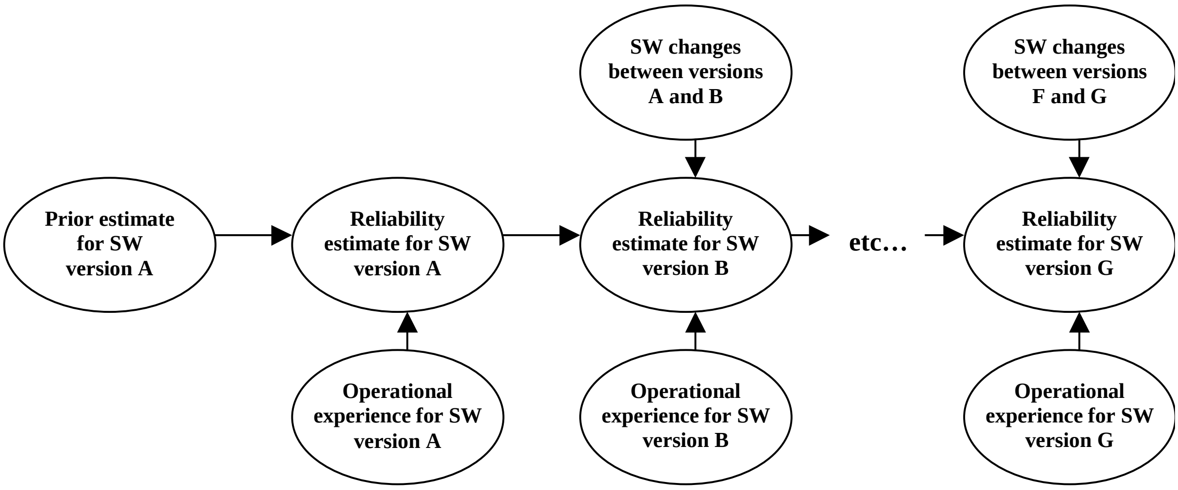

Figure 95. Diagram representing the software (SW) reliability assessment process of motor protection relay.

Calculations for different scenarios

Calculations for the motor protection relay are made for four different scenarios using all the evidence available at the time being. The scenarios differ by the assumption on the influence of software changes and by the data regarding the software faults encountered for the software versions. The different data sets are used for the conservative and neutral assumptions on the influence of software changes.

In the neutral approach the prior mean value of change in failure frequency between successive software versions was assumed zero. In the conservative approach a rather pessimistic attitude was taken, and a prior assumption was that a change in software has always a negative influence to the reliability of the system. What this means is that as a fault is removed from the software, new faults are on average introduced to the software. The magnitude of reduction in the reliability depends on the amount and criticality of the changes made to the software. In the first data set only the number of clear software faults encountered for different software versions are implemented as fault data. In the second data set the software faults and so called software inconveniences are taken into consideration.

Using all evidence in the assessment the posterior failure frequency distributions of different software versions for the conservative approach is illustrated in Figure 96. In the figure there are the posterior failure frequency distributions ranging from 2.5 percentile, the lower bar, to 97.5 percentile, the upper bar, and median marked as a dot somewhere in between. Corresponding graph for the neutral approach is given in Figure 97.

An approximation of the expectation value for the number of devices encountering a software fault during a year of operation can be obtained from the median values of distributions shown in Figures 96 and 97. The reader should, however, be reminded that median is not the same thing as expectation value even though in symmetrical distributions they concur. The posterior distributions in this case are not symmetrical and therefore median value gives only an approximation for the expectation value. For example from Figure 96 it can be estimated that for software version A it is expected that two devises out of thousand will encounter a software fault during a year of operation. For software version G the expected number is approximately four devices out of one hundred thousand during a year of operation. In the calculations all the devices are assumed to function in a similar operational profile.

Conclusions from calculations

Significant differences between the posterior failure frequency distributions of the two approaches used for the influence of software changes can’t be detected. With the conservative approach the failure frequency distributions of different software versions seem to be more monotonous, i.e. the estimates for the early software versions are better than in the neutral approach and vice versa for the later software versions. However, the difference between the two approaches for the failure frequency of the last and crucial software version is negligible as can be verified from Figures 96 and 97. Explanation to the small difference in the last software version can most probably be found from the large amount of data obtained in the assessment, and thereby the dominant role of the data in the assessment.

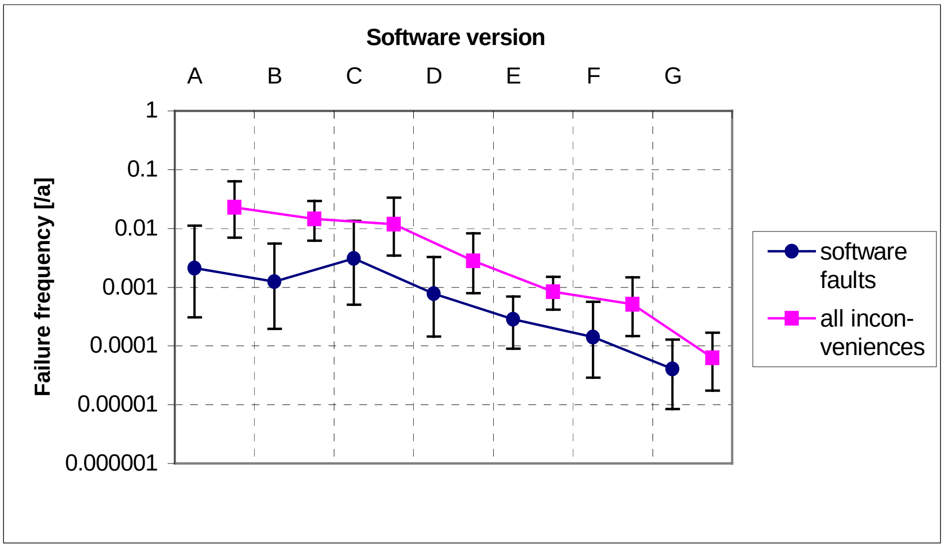

Figure 96. Posterior 0.025–0.5–0.975 percentiles for failure frequency distributions of different software versions for the conservative approach.

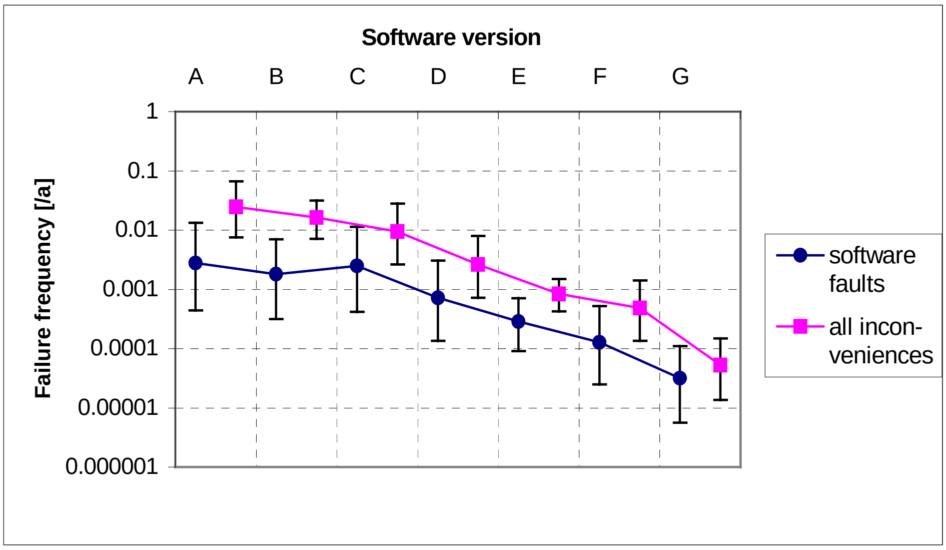

Figure 97. Posterior 0.025–0.5–0.975 percentiles for failure frequency distributions of different software versions for the neutral approach.

Conclusions

The approach used in the report for the reliability estimation of software-based system seems extremely promising. The experience and results obtained from the assessment process support our prior belief on the fact that one of the most suitable ways to estimate the reliability of software-based system involving diverse evidence from various kinds of sources is based on the use of Bayesian statistics. Especially for large systems where the system and the software contained by the system extends to the limits where the system can't be modelled or tested presumably completely it is reasonable to rely on such assessment methods as presented in this report. However, it is important to understand that the assessment method presented here utilises only part of the reliability related evidence of a software-based system and for the full comprehension of the reliability of a system a variety of different analyses such as the one shown in the report should be practised.

The assessment procedure presented enable a flexible use of qualitative and quantitative elements of reliability related evidence. At the same time the assessment method is a concurrent way of reasoning one's beliefs and references about the reliability of the system. It is therefore justified to claim that a reliability estimation method shown here can provide a strong support for the licensing process of digital I&C systems. Most of all the reliability estimation method establishes a solid ground for the communication between different participants of a licensing process.

The assessment method shown in the report is everything but complete. Further research needs to be made particularly in developing the interview process of the assessment so that the questions cover all the relevant areas of developing a high quality software-based system. The questions and interview process should be formulated so that if necessary the grounds for the answers given by an expert can be traced back to the documentation of the system. The operational experience and the operational environments where the operational experience have been collected from is also a significant area where an additional research effort is needed.

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13. Methods of risk analysis (METRI)

13.1 METRI summary report

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Abstract

METRI project focused both on probabilistic safety assessment (PSA) methodology and its applications, especially in multi-criteria decision-making situations where PSA results are combined with other, deterministic criteria to select an optimal decision alternative. The project has promoted the use of risk-informed approaches by developing methods for PSA qualification and expert panels to support risk-informed decision-making. Application areas are optimisation of in-service inspections, risk-informed safety classification of systems and components, reliability of advanced passive systems. The main topics concerning PSA methodology were model uncertainties and human reliability analyses.

Introduction

Probabilistic safety assessment (PSA) is a quantitative model to analyse the risks of nuclear power plant operation. The Finnish regulatory guide YVL-2.8 [1] states that the results of PSA must be used in support of decisions on operational safety issues. Due to improvements both in methodology and acceptability of PSA, its use in decision-making has become more and more frequent. However, as PSA is a complex model including many uncertainties and limitations, there are still research and development needs both in the PSA modelling and in the use of PSA as one decision criterion in safety related decision-making.

Main objectives

The main objectives of the METRI project were: to make progress in risk-informed decision-making methods and bring them available to end-users; develop licensing practices in problematic areas; develop methods for risk importance and uncertainty analysis. Within the risk-informed decision-making, the main objectives were to carry out application studies in the area of PSA qualification, study various probabilistic decision criteria, and review methods for pipe leakage frequency estimation for RI-ISI (Risk-Informed In-Service Inspection) applications. The studies within the development of risk analysis methods were focused on analyses of uncertainties and risk importance measures, and human reliability analysis (HRA). Application areas of HRA and uncertainty analyses were e.g. PSA level 2 issues, fire situations and passive systems.

Risk-informed decision-making

Risk-informed decision-making situations often involve experts from several disciplines. Both probabilistic and deterministic information from several sources should be combined in order to achieve an optimal decision. Within the METRI project, an **expert panel methodology** was developed to support such risk-informed decision-making processes [2]. The method was first applied in the STUK's RI-ISI (Risk-Informed In-Service Inspection) pilot study [3], where the PSA was used together with deterministic analyses in ranking pipe segments according to their risk significance. Later in the project, other applications were related to the reliability of passive safety systems and to a risk-informed classification of systems.

In the expert panel approach, quantitative risk estimates, results of deterministic calculations and other relevant information are first 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 the expert panel is to achieve a balanced utilisation of information and expertise from several disciplines in decision-making. Table 11 shows the steps of an expert panel process and the role of various participants.

Table 11. Role of participants in an expert panel process [2].

STEP	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)	familiarises 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 their 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	receives the report	comments and accepts the summary report	(comment the summary report)	summarises the discussions and results of the panel

The role of PSA in a risk-informed application should depend on the applicability of the PSA model in the decision-making situation. One achievement of the project was the development of a generic approach that would deliver a **qualified PSA assessment** over different decision contexts [4]. The main users of the developed approach are the PSA evaluators and practitioners. PSA evaluators use the approach as a decision aid for approving PSA applications in various decision contexts, e.g., related to plant modifications. PSA practitioners use the approach as a guide to modelling, analyses, documentation and quality assurance. The process for PSA qualification is illustrated in Figure 98.

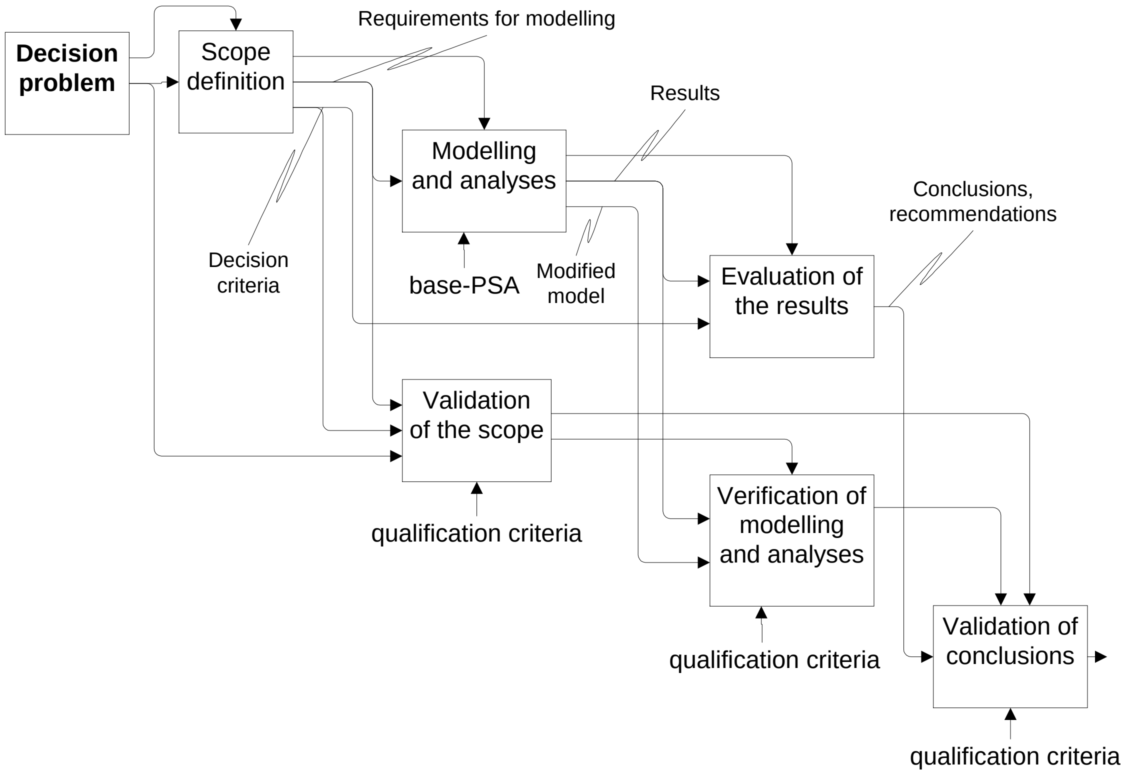

Figure 98. The PSA qualification process [4].

The PSA qualification relates rather to what would be regarded as necessary information for judging credibility of the application; it cannot ensure that the information needed is sufficient for all decision contexts. The PSA qualification thus has to represent a balanced view of the information that is provided by the PSA practitioners and needed by the PSA evaluators. This balanced view would reflect the 'best practice' in applying PSA.

A set of templates was proposed to support the PSA qualification process. The value of templates is that relevant items to be accounted in a PSA-application are summarised in one document. Templates should provide the necessary and sufficient assurance of the quality of the application. In practice, each utility would have to design PSA qualification templates according to their own PSA-practices.

One possible application of risk-informed decision-making is the consideration of PSA results in determining the safety classification of components, systems and structures. The basic idea of this **risk-informed graded QA** is that the quality assurance requirements should be consistent with risk importance. So far, the safety classification has been based on deterministic safety analyses, and it is not always in accordance with PSA results. A study on the applicability of a risk-informed graded QA [5] is summarised in a separate article of this FINNUS final report.

Along with the adoption of risk-informed decision-making principles, the need for formal **probabilistic decision criteria** has been risen. However, there are many practical and theoretical problems in the application of probabilistic criteria. One has to think what is the proper way to apply probabilistic rules together with deterministic ones and how the criteria are weighted with respect to each other. These questions were addressed from a decision theoretic point of view [6]. A decision analytic framework for evaluating the criteria was presented, and various criteria were studied to demonstrate their behaviour under incompleteness or uncertainty of the PSA model. Principles to evaluate and apply various decision criteria were identified and reported, and recommendations were given on the application of the criteria in different decision situations.

System analytic approaches to **manage ageing and maintenance** were demonstrated within the project. The work was related to optimisation of in-service inspections by taking into account PSA results, and to maintenance strategic classification of equipment.

In the **risk-informed in-service inspection** (RI-ISI) methodology, the information from PSA is used to identify the safety significance of piping segments, and this information is combined with the degradation and failure potential to obtain a risk ranking of piping. The applicability of the RI-ISI methodology was addressed both considering the consequence and failure frequency evaluation. The previously described expert panel approach was used within STUK’s RI-ISI pilot study to combine the deterministic information on degradation mechanisms and probabilistic information on pipe break consequences.

One topic of interest is the determination of degradation potential of piping. In the pilot study, the evaluation was done qualitatively, but later on the evaluation of quantitative approaches has been considered. In this connection, existing probabilistic fracture mechanics (PFM) models used in RI-ISI applications were reviewed. Further, a comparative study of two approaches to estimate leak and rupture frequencies for piping was conducted [7]. One method is based on a PFM model while the other one uses statistical estimation from a large database. The study identified large differences in the rupture frequencies obtained by the two alternate approaches. The rupture frequencies calculated by the PFM codes have larger variation, which is understandable since the models use plant specific information on stresses that may vary quite significantly. Differences may be explained on one hand by different expert judgements, on the other hand by different possibilities to account e.g. for weld-specific information.

The information based on statistical evidence should be taken into consideration, e.g. in updating of LOCA frequencies. For this purpose, the principles of the statistical approach might be sufficiently accurate, especially if it is completed with structured expert judgement. The PFM approach requires a lot of weld specific information from the stresses, and it may be argued that the approach may be too laborious for some applications. The advantage of the PFM model is the explicit treatment of inspection reliability, which enables sensitivity studies with different inspection policies. As the primary interest in RI-ISI is in the risk ranking of the welds, the absolute quantitative results are of less importance than the relative results, which are not sensitive for eventual conservative assumptions of the model.

A survey on principles and examples of **maintenance strategy classifications** of equipment and systems at Barsebäck, Loviisa and Olkiluoto plants was done as a Nordic co-operation [8, 9]. The survey compares the decision objectives and criteria used in maintenance planning. Maintenance strategy classification aims at ranking the plant items for the allocation and planning of maintenance actions and related resources. Another aim is to focus the maintenance analysis work to the most important maintenance items. The next step in the maintenance strategy development is to optimise the maintenance by enhancing actions of important plant objects and reducing maintenance costs on equipment of less maintenance importance. A selective and experience-based reliability centred maintenance (RCM) analysis of the bulk of the failure and maintenance history data is recommended to support the optimisation.

Development of risk analysis methods

Almost all advanced NPP designs include passive safety systems. However, the reliability analysis and licensing practices of such systems have not been studied sufficiently, as concluded in an international inquiry in OECD countries carried out as a part of the METRI project. The activities within METRI have also included a review of existing approaches to **passive system reliability methods**, developing classifications for different passive systems and preliminary views for their licensing practices [10]. Further, methods have been developed for qualitative uncertainty analysis and they have been applied in an analysis of emergency condenser of an SWR reactor [11]. An expert panel was conducted with STUK to form interdisciplinary views about reliability aspects of passive systems. The panel consisted of experts in PSA, thermohydraulics and processes.

The METRI project considered both qualitative and quantitative **uncertainty analyses** related to PSA. A qualitative approach to identify and summarise major uncertainties related to an analysis of a phenomenological issue was developed [12]. In this approach, a specific form was used to document major uncertainties, their nature, importance and possible means for uncertainty reduction. The approach should improve the uncertainty communication between analysts representing different expertise and decision-makers. The approach and the uncertainty documentation forms were tested in case studies. One of them was related to a BWR reactor building hydrogen combustion scenario that was analysed within the MOSES project. Other case studies were related to a fire PSA application and analysis of uncertainties of a passive safety system. Regarding quantitative uncertainty analysis, a review of probabilistic sensitivity analysis methods was made.

The work concerning **importance measures** was concentrated on reviewing and generalising structural importance measures [13]. These measures are based purely on the redundancy structure of the systems, and the information on components' failure probabilities is not taken into account. The importance measures proposed in the study can be determined on the basis of minimal cut-sets of the system. The structural importance measures can be applied e.g. in refining safety related technical specifications. In addition to this, some recommendations for applying and presenting the traditional importance measures were made in a separate study.

Another issue related to PSA-importance measures is the analysis of significance of physical phenomena in the PSA-level 2 models. This kind of analysis aims at identifying most important phenomena having impact on the probability of large releases and at defining possibilities to reduce uncertainties. The approach taken in the project in resolving this problem is based on using the results of Monte-Carlo simulations of the level-2 PSA models and defining suitable phenomenological importance measures. The approach is being applied in a small case study.

Research efforts in the area of **human reliability analysis (HRA**) were focused on developing methodology for analysing human errors of commission. Another topic considered was the integrated sequence analysis, where interdisciplinary approaches were developed and applied to analyse fire situations.

The framework for analysing human commission errors, i.e. wrong human actions, was developed in the project [14]. This framework, called FACE, consists of five phases: 1) target selection, 2) identification, 3) screening, 4) modelling and 5) probability assessment. Definition of the context and factors influencing human performance are an important part of the approach. For the identification phase, checkpoints and guiding questions have been developed to help the analyst.

The FACE method has been applied to analyse a disturbance at Loviisa NPP. In this scenario, a spurious “high pressure in containment” protection signal was considered and the possible operators’ actions were studied. Decision points and consequences were identified and probabilities evaluated. Another application was related to the analysis of severe accident management action at Olkiluoto NPP, namely a lower drywell flooding. The objective was to gain experience from applying FACE-method and analysing human reliability related to severe accident management actions, in particular related to actions taken into account in level 2 PSA. The study supports the review of HRA for level 2 PSA [15].

Errors of commission were also addressed within a Nordic study [16], where operating experience related to wrong human actions from several Nordic NPPs were collected and analysed. This study covered not only control room operations but also maintenance, testing and outage management actions. Figure 99 shows a barrier model illustrating how causal factors lead to failed human activities, and result in consequences due to the fact that barriers fail. Finally, the progression of the event is stopped by an efficient barrier function, which may be physical barrier, a safety system or an organisational function. Although the proportion of wrong human actions was high in the analysed material only few actions led to wrong system functions and disturbances. A significant number of events were due to human actions outside the control room, i.e. in maintenance, testing and operating actions all over the installation.

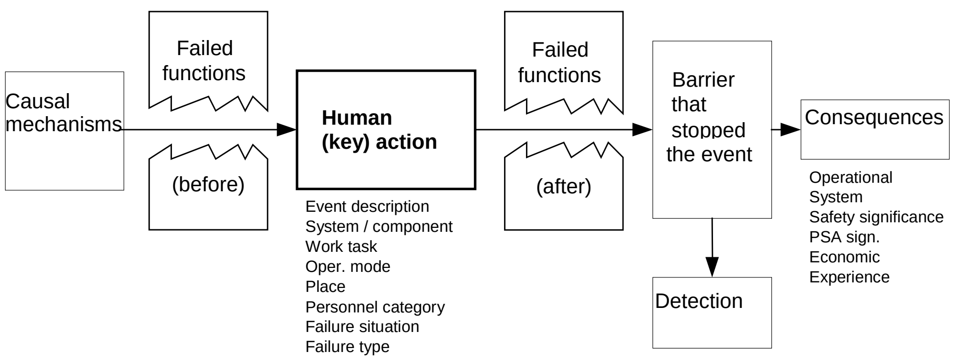

Figure 99. A schematic representation of the classification used in the study of commission errors and the location of various categories in a barrier model [16].

Human reliability analyses have also been conducted in co-operation with the FINNUS/WOPS and FINNUS/FISRE projects in the area of fire situations. The aim of this co-operative study is to produce information to improve the realism of human reliability analysis methods. The results will be of practical use in improving the nuclear safety related to fire situations. The study integrates the psychological and the decision analytic approaches, and uses the experience obtained from fire research at VTT. The study is summarised in a separate article of this FINNUS final report.

Applications

The results of the METRI project have direct applications in risk-informed decision-making both at nuclear safety authority and utilities. As summarised in the previous section, several research and development studies have included a practical application to test the developed approaches. The expert panel approach was applied in three separate cases: pilot study of risk-informed in-service inspection process, graded QA taking into account PSA results and passive system reliability. The feedback from the expert panel has been very positive. Within the development of the approach for PSA qualification, the proposed templates were applied to practical cases at the utilities. The experience from these case studied was that the qualification templates would presumably enhance the quality of documentation of PSA applications.

The developed human reliability analysis methods were also applied in practice. The methodology to analyse errors of commission was applied to different cases related both to control room operations and to human actions outside the control room, e.g. maintenance related errors. The analyses of operating experience from commission errors at several plants resulted in findings regarding type of errors that could occur and human interactions that should be considered in PSA. The method developed for the analysis of requirements and action possibilities in fire situations can be applied in the training of operators as well as in the development of operating procedures for fire situations.

Conclusions

Along with the increasing use of PSA in safety related decision-making, the development of decision analytical methods and tools to facilitate the risk-informed regulation and safety management has become important. The integration of different expertise in the decision process calls for formal approaches, in order to achieve a balanced result. It is also important to verify the quality of PSA in relation to the decision in question, so that an appropriate weight can be given to the PSA results in risk-informed decision-making. Although the PSA models are very advanced, there are continuous development needs in many areas, e.g. in human reliability analysis. The

METRI project has addressed topics related to reliability analysis of passive systems, analysis of human errors of commission, and methods for uncertainty analysis.

As the risk-informed decision-making is a very interdisciplinary subject, the METRI project has promoted the co-operation between other projects in the FINNUS-programme. Within human reliability analyses joint studies have been conducted with the WOPS and FISRE projects. Co-operation with the AGE, INSMO and STIN projects has been done in the area of structural reliability and risk-informed in-service inspections. In relation to passive system reliability, connections have been established with the TOKE project, and phenomenological uncertainties have been considered together with the MOSES project. The METRI project has also been active in international co-operation through e.g. OECD, NKS and ESReDA. The results of the project have been presented internationally at the most important conferences in this field.

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13.2 Risk-informed graded quality assurance

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Abstract

The grading of quality assurance (QA) controls at nuclear power plant aims at assuring an acceptable safety level but also at allocating QA resources cost-effectively. Historically, QA requirements stem from the safety classification of the systems, structures and components based on deterministic safety criteria. Risk-informed approaches applying the probabilistic safety assessment (PSA) are considered as one solution to rationalise and potentially relax the QA system. In this paper, risk-based conditions are formulated for the assignment of QA grades. The risk-based formulation is a reference model to explore possibilities, requirements and obstacles of this application of PSA.

Introduction

In nuclear power plants, the systems, structures and components important to the safety are classified according to their safety significance. This safety classification determines requirements in design, qualification and regulatory review. It also implies the strictness of quality assurance (QA) controls that shall be followed in design, manufacturing, installation and operation of the items. QA requirements shall be consistent with the item's importance to nuclear safety [1]. This grading of QA controls aims at assuring that acceptable safety level is achieved but also at cost-effective allocation of the QA resources. The critical questions of the safety classification and graded QA system are:

1. How should the safety significance of an item be assessed?
2. How do a safety class and associated QA controls affect the reliability of the item?
3. Provided that we have method and data for the assessments 1 and 2, how should a safety class and associated QA controls be assigned for a particular item?
4. Provided that we have a decision rule to select a safety class and grading of QA controls for one item (solution to the question 3), how should the quality system be optimised as a whole, e.g., with respect to allocation QA resources and resolution level of safety classes and the quality system?

Historically, the assessment of safety significance of items of nuclear power plants has been based on general design criteria and on deterministic safety analyses. This assessment has been followed in the safety classification of the items. QA grades for organisational functions (e.g. maintenance, training, preparation of instructions) have been assigned more or less strictly according to the safety classes, at least for the most safety significant classes.

Today, the operating nuclear power plants have experience that their safety classification system and QA requirements are partly unbalanced and inflexible. For instance, PSA can be used to identify deviations between risk importance ranking and safety classes [2]. Low risk-significance could indicate candidates for reductions in QA requirements. *Risk-informed* approaches based on PSA are considered as one solution to rationalise and potentially relax the QA system. PSA provides an approach to assess the safety significance (question 1). PSA also touches the question 2 when assessing the reliability of the items. There are, however, many questions to be resolved for a full application of risk-informed decision-making to solve problems 3 and 4.

This paper discusses the use of PSA for the assignment of safety classes and QA grades [3]. *Risk-based* conditions are formulated for the assignment of QA grades as a function of PSA results and other decision criteria. The risk-based formulation is a reference model to explore possibilities, requirements and obstacles of this application of PSA.

The relationship between risk and QA grades

In the graded QA philosophy, the assignment of QA control and verification measures for an item can be represented as a problem to determine the appropriate quality grade q among the set of grade options {QA1, …, QAn}. The grade determines the practical control and verification measures described in the QA programme. The grades are ordered with respect to their strictness: QA1 $>_q$ … $>_q$ QAn, where "$>_q$" is interpreted as "is more stringent QA grade than." Ideally, by choosing a more stringent QA grade, the reliability of an item can be improved. This condition can be expressed as follows

$$q_1 >_q q_2 \Rightarrow p(q_1) \leq p(q_2), \tag{20}$$

where q_1 and q_2 are alternative QA grades, and $p(\cdot)$ represents the relationship between the item's QA grade and unavailability.

Subsequently, we assume that by improving the item's reliability (decreasing the unavailability), we can reduce the core damage frequency, as follows

$$p(q_1) < p(q_2) \Rightarrow f(p(q_1)) \leq f(p(q_2)), \tag{21}$$

where $f(\cdot)$ is core damage frequency as a function of the item's unavailability.

The conditions (20) and (21) are desirable properties when using QA grades as a method to control safety of nuclear power plant. Figure 100 illustrates the situation.

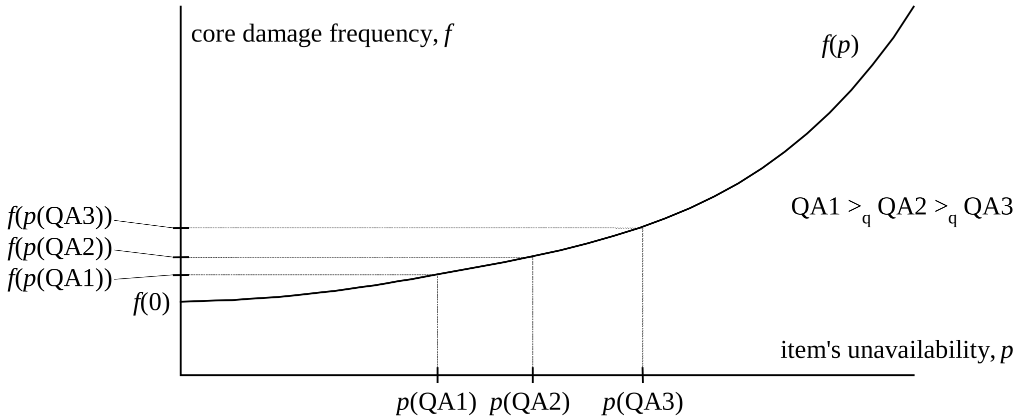

Figure 100. The ideal relationship between QA grades, an item's unavailability and the core damage frequency. More stringent QA grade implies lower unavailability that implies a lower core damage frequency. f(0) corresponds to the case when the item is totally reliable, i.e., p=0. The slope of f(p) is item-specific.

The formulations above can be generalised to multi-dimensional cases: (1) there are more than one risk measure of interest associated with the safety objectives, (2) there are several QA measures affecting reliability of the item, (3) there are several reliability parameters associated with the item, affected by QA measures, affecting risk measures. The conditions (1) and (2) are similar and have the same interpretation [3].

Risk-based assignment of the QA grade using the ALARP-principle

As a basis for supporting safety related decision making, we adopt here the ALARP-principle (As Low As Reasonably Practicable). The risk of a system is divided into three parts: the unacceptable risk region, ALARP region and negligible risk region [4]. The ALARP-principle assumes a level of risk that is tolerable to the public, e.g., the maximum tolerable core damage frequency f_{high}. The risk posed by any new system shall at least be below that level. There is a second level of risk so low that the public will accept that "it’s not worth the cost" to reduce it further, e.g., the negligible core damage frequency f_{low}. Between these two risk levels there is the ALARP region, in which the cost associated with the system change option is compared with the amount of risk reduction achieved.

The ALARP-principle can be applied also in a relative manner. Then we compare the change from the present risk level, Δf, with the criteria Δf_{high} and Δf_{low}. The benefit of the Δf_{high} -criterion is that it is more robust with respect to uncertainties of PSA than the

f_{high} -criterion. The Δf_{low} -criterion can be applied to argue that the item has negligible importance to safety, and thus the item's QA-requirements could be relaxed or removed.

In practice, combinations of absolute and relative criteria are applied. Following the ALARP-principle, the f_{high} and Δf_{high} -criteria are boundary conditions to reject unacceptable options. In the ALARP-region, a preference model is defined for ranking of the options, e.g., an additive multi-attribute value/utility function

$$\min V(q) = \Sigma_i \, w_i \, v_i(q), \text{ so that} \tag{22}$$

$$q \in \{\text{QA1}, \ldots, \text{QAn}\} \tag{23}$$

$$f(p(q)) \leq f_{\text{high}} \tag{24a}$$

$$f(p(q)) - f(p(q_{\text{ref}})) \leq \Delta f_{\text{high}}, \tag{24b}$$

where $V(\cdot)$ is a value function, v_i, i = 1, …, the attributes of the preference model, w_i:s the corresponding weight factors, and q_{ref} the present QA-grade. Weight factors encode the trade-offs that we are prepared to make between the attributes. The minimisation amounts to choosing the quality grade of an item that is the least stringent, but acceptable according to the ALARP-principle.

The critical steps of the structuring the preference model are to define the attributes and to determine the weights. In the cost-benefit analysis, we apply two attributes: the core damage frequency and cost of the QA-grade. Weights stand for the price of the risk increase/decrease.

Practical and theoretical considerations

Principles and conditions (20)–(24) are theoretical and subject to several difficulties in practical applications but also theoretical weaknesses are included in them. The following questions could be raised against the risk-based approach:

- How to assess the dependence between the item's unavailability and QA grade? There is limited if any data to estimate the form of $p(q)$. This problem is specific for the graded QA application. The US NRC's (Nuclear Regulatory Commission) guide recommends performance of bounding analyses for checking purpose [5]. IAEA's Technical Document presents a method for qualitative rating of maturity and complexity of the item during each life cycle phase [6]. Maturity and complexity -attributes implicitly reflect the sensitivity of the item with respect to the QA grade.
- How to deal with uncertainties of PSA? How to deal with an incomplete PSA-model? This is a generic problem for all PSA applications. The role of uncertainties of PSA can be analysed, e.g., by means of uncertainty and sensitivity analyses. The same could be asked for deterministic approaches.

- Which risk measure(s) should be applied to the assessment of the safety significance? PSA provides core damage frequency and large early release frequency. Fussell-Vesely and risk increase factor importance measures are usually applied to measure the safety significance. These are presumably enough. However, guidance is needed for use of several measures in parallel. See discussion and examples in [7].
- How to determine ALARP-levels of maximum tolerable and negligible risk level? Who determines the ALARP-levels? In many countries, safety authorities have defined explicit risk criteria. Quantitative risk criteria are also presented in many international standards and guides. A typical maximum value for the overall core damage frequency is 1E-4/yr, see e.g. [8].
- Should we use absolute or relative risk criteria or both? If both, how? This is a generic problem of PSA applications and discussed e.g. in [7].
- What kind of preference model should be used in the ALARP-region (equation (22) is a simple additive linear model)? How should weights be assessed and who should assess them? Is it enough to consider risk and cost or should we have other decision criteria? French *et al.* [9] argues that multi-attribute utility methods are more satisfactory way than the (two-attribute) cost-benefit analysis.

Discussion and conclusions

In USA, there is a strong support to use risk-based insights to complement the earlier deterministic rules. The NRC encourages the utilities to use PSA to improve safety decision making. The NRC has prepared regulatory guides for a number of application areas, including risk-informed graded QA [5, 8]. PSA-evaluation of safety significance of the items is just one step of the application. The emphasis is put on the integrated assessment of the safety significance of the item based on traditional engineering, probabilistic and qualitative information available. The integrated assessment is performed by an expert panel.

Representatives of the Finnish nuclear safety authority and utilities have been interviewed to assess the interest for risk-informed graded QA [3]. Concerning the application in the present nuclear power plants, the general attitude is reserved. There is no desire to complicate the safety classification system with an additional risk-based classification. PSA insights could be applied to change safety classes of certain systems and items. On the other hand, PSA is already used in many applications related to the QA-system, e.g., optimisation of test intervals and in-service inspection strategies. The utilities apply risk importance measures to classify and rank the items for maintenance planning. Concerning new plants, common opinion is that PSA-insights should be utilised for the determination of safety classes and QA grades.

The fundamental problem of risk-informed decision making models is the ambitious goal to capture all risks, uncertainties and values into a sound mathematical model. The extent of facts, judgements and assumptions in the PSA-model is beyond the comprehension of non-PSA-experts. The subjectivity of (any) decision-making model is difficult to accept by outsiders. These obstacles should not, however, discourage us from developing risk-informed approaches for safety management. PSA could be made more transparent and pedagogic; use of preference models to support decision-making should be exercised. For risk-informed graded QA, data is needed to evaluate the effect of QA controls on an item's reliability. If we think that questions 1–4 posed in the introduction are relevant, we should seriously consider risk-informed graded QA.

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

14.1 WOPS summary report

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Abstract

WOPS project concentrates on human and organisational factors. The project focused on two main topics, the development of the *competencies of control-room operators* and the *organisational culture in high-reliability organisations*. Within these topics several studies were accomplished in which practically relevant problems of the both Finnish Nuclear Power Plants (NPP) were tackled. Thus, the studied issues were transformation of operational expertise in generation change situation, constraints on control-room operators in fire situations, organisational culture in the STUK Nuclear Reactor Regulation department and organisational culture in the maintenance department of the Loviisa NPP. As a background of the latter study an analysis of human errors in maintenance was conducted. New empirical results were achieved in these studies. The work also contributed to a long-term methodological aim to develop a new framework for the analysis of human conduct in dynamic, complex and uncertain situations.

Introduction

As a part of FINNUS program WOPS (Working practices and Safety culture in NPP operations) project concentrates on human and organisational factors. A prerequisite for safe and efficient operation and maintenance of nuclear power plants is the understanding of the regularities of human activity. Knowledge of everyday practices and especially experiences of disturbance situations and accidents have shown that human activity plays an important role in safety critical processes and in knowledge intensive work. Questions like, how to maintain competence through the change of personnel generation, how to learn more from human errors, how to evaluate and develop organisational culture, and how to improve the management and co-operation of fire situations, represent nowadays challenges faced in Finnish NPP industry and authority.

In concert with many our colleagues who study human behaviour in complex work domains, for example in aviation, NPP or the medical domain, we have identified the need to develop new methodological approaches. The toothlessness of available methods becomes visible in incapability to rise to the challenges of practical human factor issues. Our work for designing a new framework was initiated in the previous research programme on nuclear safety. In the present programme this work was continued and a novel research methodology is now emerging. As an ecological methodology our framework conceptualises human action as a continuous interaction between the human and the environment. The interaction is structured in accordance with the anticipated results. These promote possibilities for the survival of the unified human-environment system. An ecological approach is often articulated as practically relevant but there is still a great demand for empirical research methods for realising the ecological principles. Our work has oriented towards producing such empirical methods through studies that we have conducted in the field together with domain experts and the actors themselves.

A further qualifier of our methodology is that it considers human activity as intentional, context-dependent and situationally constructed enterprise. This implies that causal explanation, that provides a proper frame when investigating connections between events in the physical world, must be completed with understanding what behaviour is a sign of, what it means. Thus, in a truly psychological analysis of behaviour, the starting point must be the subjects' own accounts of their actions. The methodology under development was in its earlier phase called the contextual analysis of working practices. We developed a practical application of this methodology for the control room operators' simulator training [4] [5]. The method was called TAPA (habit), a name that anticipated the further theoretical developments. During the present research programme we have elaborated the theoretical and methodological basis of the framework and have coined our methodology the Core-Task Analysis (CTA) [2][3]. We can distinguish two factors that gave impetus for the further development of the method. Firstly we realised the need to consider work as it is in normal practices. While work practices may be shaped by training and exercising with simulators, they are still formed in the normal daily work under the influence of the organisational practices and management policies in the plant. There is very little knowledge of the cognitive, emotional and cultural mechanisms of this development process and their interaction with the formation of professional skills. The second springboard for the development of the methodology was the discovery of the theoretical possibilities that the American pragmatist tradition and especially its concept habit had to offer for the explaining of the construction of actions in daily practice [3].

WOPS project focused on two main themes, which are both theoretically and practically important issues in NPPs. The first is the competencies of the "sharp-end" actors, the control room operators. Knowledge of the tasks and qualifications of operators is necessary for any intelligent development of information tools in the control room, or for efficient design of training. The other main theme was organisational culture in high-reliability organisations. This theme highlights our conception that actions of the personnel are shaped within a shared organisational context that provides the actions with their meaning. During the four-year project these main themes were tackled through five sub-projects.

The first sub-project focused on the *transformation of expertise* from the older to the younger operator generation. The second project referring to operator competencies focused on understanding the *task demands put on control-room operators in fire situations*.

The studies on the organisational culture of high-reliability organisations focused first on the *organisational culture of the nuclear authority organisation.* This study lead to the next study, *organisational culture in the maintenance organisation of a NPP*. Human error studies have demonstrated that maintenance errors are a significant source of unreliability in the NPPs. Also in our project a comprehensive analysis of *human factors in maintenance* was conducted and it provides an important basis for subsequent organisational study.

Each of the five sub-projects will be described shortly in this project summary. The present volume also includes more extended papers of two studies, the fire situation study and the organisational culture studies.

Participation in two European research networks gave an international background and connection to our work. The first European project LEARNSAFE focused on organisational culture, and the second on skills, competencies and learning in knowledge-intensive work Work-Process Knowledge (WHOLE).

Main objectives

The aim of the project was to enhance safe operation and maintenance of nuclear power plants. Based on analysis of human decision-making, conceptions of work, action in actual working situations, literature reviews and theoretical work the project aimed at the construction of methods for the evaluation and development of working practices and organisational culture in the plants.

There were several methodical challenges in the project. The first problem was how to study organisational culture contextually. Related with this, our aim was to complete the behavioural analysis of working practices within the Core-Task Analysis with an organisational approach to practices. Such an analysis could reveal the culturally shared meanings and values underlying actions. It was, secondly, found necessary to complete the analysis of practices through conceptualising the emotional-energetic aspects of work and expertise. Thirdly we wanted to continue the development of the modelling of the intrinsic constraints of different domains and situations on action. Such modelling of the environmental features is an important ingredient of the ecologically-oriented Core-Task Analysis. The modelling of requirements and action possibilities in fire situations was considered a practically important object of study.

A further aim was to carry out international co-operation with relevant partners within the NPP and the cognitive ergonomics community. The project also aimed integrating the research results in the on-going human factors development activities. Moreover, an aim was to promote development of competencies of young researchers in the field.

Operators' competencies and transformation of expertise

The study on transformation of expertise in generation change produced several theoretical and methodological results. The operator actions have predominantly been conceived as actions in disturbance situations. The question of the requirements and possibilities for developing expertise through experience in normal work has seldom been asked. We approached this topic through utilising the CTA framework. The core task refers to the essential content of the work that has to be fulfilled in every situation. Core task analysis is seen as a tool for reflecting current practices and developing new ones through enabling to direct the operators' attention at the essential content of work as it appears in concrete situations. The shared reflection and communication of practices is the means for transmission of skills and knowledge to novices. Through new interview material, further analysis of previously collected material, and theoretical work we were able to deepen the conception of nuclear operators' core task. An important result was that more attention could be paid at the subjective and emotional aspects of behaviour in the situational organisation of actions, regarding particularly the mobilisation of personal energetic resources. The result was a motivational performance model (MPM) [7]. The central concept in the model is the expert identity. This concept refers to the emotional aspects intertwined with work performance, especially coping with demanding situations, and the development of expertise in work.

Based on interviews on development of professional competence challenges of the change of generation for maintaining operator competencies at the nuclear field were clarified. Interview material was analysed and a model of the development of professional competence of the operator trainees was created. The training and socialisation practices of the target organisation were identified. The operator's conceptions of the strengths and weaknesses of the training of newcomers at the target organisation were identified and presented to the organisation. Training practices were compared to contemporary theories of learning and some discrepancies were found. [6]

Modelling the constraints of fire situations

In this study the aim was to develop a method for analysing and modelling requirements and action possibilities in fire situations. The method was intended to support the decision-making of the control room personnel and the co-operation of the control room, the fire fighters and the security guards.

A method for analysing and modelling requirements and action possibilities in fire situations has been developed. The method was tested in some cases of fire situations. Expert interviews were carried out in order to gain knowledge of characteristic features and problems of NPP fire situations and of problems of probabilistic fire safety assessment (fire PSA). Another topic considered was the integrated sequence analysis, where interdisciplinary approaches were developed and applied to analyse fire situations. Human reliability analyses have also been conducted in co-operation with the METRI and FISRE projects in the area of fire situations.

The aim of this co-operative study was to produce information to improve the realism of human reliability analysis methods. The results will be of practical use in improving the nuclear safety related to fire situations. The study integrated the psychological and the decision analytic approaches, and used the experience obtained from fire research at VTT. The method developed for the analysis of requirements and action possibilities in fire situations can be applied in the training of operators as well as in the development of operating procedures for fire situations.

Organisational studies at STUK and NPP maintenance

In our studies on organisational culture the concept ”safety culture” has been specified and a methodology to study safety culture within high reliability organisations has been proposed. In addition to the development of the methods, the empirical studies conducted in the project provided both general models and specific results of the target organisations. In the organisational culture theme a case study was conducted at STUK’s Nuclear Reactor Regulation (YTO) – department and the other in Loviisa maintenance organisation [8] [9]. A model of the demands of the regulatory culture and regulatory practice was proposed. Results of modelling the maintenance task in nuclear power plant show that there are three critical functions to be controlled in all different levels in organisation. These are anticipation, reacting to deviation recognised and reflecting the accuracy of actions and decisions. These functions set different demands to maintenance organisations practices and resource allocation. A survey of the organisational culture and core task of a maintenance department of a nuclear power plant was conducted as a part of the case study.

Focusing on maintenance gained motivation from the international and Finnish experience that indicates that maintenance activities are critical for the safe and efficient operation of the NPPs. Human error studies in maintenance have demonstrated the importance of human factors for the reliability of maintenance [10]. A comprehensive analysis of common cause failures was conducted in the sub-project *human factors in maintenance*. The study aimed at conceptualisation of the maintenance task, its goals, critical functions and the demands for the actual organisation of the maintenance. Under this theme the existing experiences and research findings regarding human reliability of maintenance of the Finnish NPPs were studied, and the needs for further human factors work in the area were conceptualised.

In the WOPS project several literature reviews were conducted. They provided information about theories and main concepts of organisational culture, learning in work and summarised research results about the effects of NPP fire situations on control room information presentation and the operators' possibilities to control the process. Moreover, the literature review on human factors in maintenance showed that international research on maintenance work in safety critical industry was both heterogeneous and few in number.

As a part of international co-operation in the project interviews on quality assurance were made at all Finnish and Swedish nuclear power plants and at the OECD Halden Reactor Project.

Applications

The project provides information for managing three practical aims: development of human-machine interface, development of competence and training and development of organizational practices. The framework, the developed methods and models can be applied in these three areas. The work done in the project can be applied to support training, development of instructions and fire PSA.

In order to ensure the straight impact on NPP safety and get projects results into practice seminars and work groups were organised in co-operation with NPP personnel, STUK personnel, NPP experts and researchers. The empirical results and development activities conducted in the case studies contributes for the improvement of nuclear safety.

Conclusions

Development of complex work and the information technological artefacts for its mastery is a continuous process that assumes interdisciplinary research and development and close connection between theory and practice. On the basis of our studies we see three major areas that deserve attention in the future development of the safety in the present and the possible future nuclear power plants.

The basis for any developmental actions in work is a profound understanding of the requirements of the task and the nature of expertise needed for the mastery of the work. The Core-Task Analysis methodology has deliberately been designed to support the enhancing of such understanding. The presently on-going technical modernisation of the plants induces major changes in the human-environment interaction. Before the new technologies are installed there is a need to analyse and anticipate the effects of the new artefacts on the co-operative and operational tasks of the operators. Such analyses may serve as background for design and training solutions.

The design of the human-technology interaction in the control rooms and in other work situations is the second major research challenge. The adoption of a human-centred design is needed for intelligent exploitation of the possibilities of the modern information technology. The current usability and interface conceptions are mainly developed for the design of smaller products and consumer goods. Therefore there is a need to develop the conceptions of the usability of interfaces to embrace deeper functionality demands on the systemic information artefacts. Interdisciplinary research including industrial designers, domain experts and cognitive ergonomics should be promoted. Such research should not restrict to the analysis of the products only but it should try to develop means to create insight and empirical knowledge of the design process itself.

A further change factor in the present situation is the approaching generation change in the plants. These changes set forth a major challenge because they touch upon the whole activity system, from operating, maintenance, engineering and design activities up to the management and regulatory control of nuclear power production. We see that the change may be mastered the better we understand the functioning of the organisation and the factors that enhance and limit its development. Pertaining to this it is necessary to analyse the meanings and values that people in the organisation seem to hold. These factors constitute the organisational culture of the work place and they shape the behaviour and decisions of the actors in particular work situations. Thus, analysis of the organisational culture, the working practices, and technical artefacts, constitute three completing aspects to understanding activity in an organisation.

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14.2 Contextual assessment of organisational culture – methodological development in two case studies

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Abstract

Despite the acknowledged significance of organisational culture in the nuclear field, previous cultural studies have concentrated on purely safety related matters, or been only descriptive in nature. New kinds of methods, taking into account the overall objectives of the organisation, were needed to assess culture and develop its working practices appropriately. VTT developed the Contextual Assessment of Organisational Culture (CAOC) methodology during the FINNUS programme. The methodology utilises two concepts, organisational culture and core task. The core task can be defined as the core demands and content of work that the organisation has to accomplish in order to be effective. The core task concept is used in assessing the central dimensions of the organisation's culture. Organisational culture is defined as a solution the company has generated in order to fulfil the perceived demands of its core task. The CAOC-methodology was applied in two case studies, in the Radiation and Nuclear Safety Authority of Finland and in the maintenance unit of Loviisa NPP. The aim of the studies was not only to assess the given culture, but also to give the personnel new concepts and new tools for reflecting on their organisation, their jobs and on appropriate working practices. The CAOC-methodology contributes to the design and redesign of work in complex sociotechnical systems. It strives to enhance organisations' capability to assess their current working practices and the meanings attached to them and compare these to the actual demands of their basic mission and so change unadaptive practices.

Introduction

Psychological studies in the nuclear field have mostly concentrated on process control, which is considered as complex sociotechnical activity. Vicente [25] characterises complex sociotechnical systems as e.g. consisting of many coupled subsystems, having uncertainty in data available to workers, having mediated interaction via various tools and having potentially high hazards ([25], pp. 14–17, see also [16]). These characteristics are not unique to process control, but can be considered as applying to many organisations and functional units operating in the nuclear field, e.g. regulatory authorities and maintenance organisations. These complex systems can also be conceptualised as *organisational cultures*.

In the nuclear field, a term *safety culture* [8] is frequently used to describe attitudes (and values) required for a reliable and safety conscious worker and employer. The attitudes emphasise questioning, rigorous and prudent approach and communication as a basis for sound safety culture. [8] Safety culture concept is often presented separately from the organisation's other characteristics, such as organising of work, technology, organisational structure, business strategy and financial decision-making. Safety culture is thus considered to be independent of (or only loosely dependent on) the wider organisational culture. This conceptual separation easily reduces the term safety culture to refer only to factors that are known in advance and are clearly connected with safety, such as safety attitudes and safety values. This results in the loss of the holistic perspective originally sought with the organisational culture concept. [21] Pidgeon [17] cites Kennedy and Kirwan [9] who conclude that "the existing attempts to study safety culture and its relationship to organisational outcomes have remained unsystematic, fragmented, and in particular underspecified in theoretical terms [9]".

On the basis of theoretical and empirical work in the WOPS project of the FINNUS research programme (see e.g. [18], [19], [21], [14]), we propose four primary reasons to use the concept of organisational culture instead of safety culture in research and development work at high reliability organisations:

A) Organisational culture consists of values, conceptions, assumptions and meanings of the personnel, which manifest themselves in e.g. organising of work, utilisation of available tools and resources and working practices, and hence has an influence on the effectiveness of the organisation ([24], [21], [1]).

B) Organisational culture has a tendency to resist change. Changes in the operating environment do not automatically reflect to organisation's practices. [24]

C) Organisations and communities of practice [10] can develop unadaptive and narrow routines, strong preconceptions [1] and working practices as characteristics of their culture. These might no longer serve the core task [13] and could be experienced as unmotivating.

D) Since the culture inherently resists outside influences but is not static (it develops constantly), the only way to develop it in the appropriate direction is to analyse the dynamics of culture and tailor the developmental initiatives to the capabilities of the culture in question. [24], [21]

Despite the acknowledged significance of culture in the nuclear field, previous cultural studies have concentrated on purely safety related matters, or been only descriptive in nature. New kinds of methods, taking into account the overall objectives of the organisation, were needed to assess culture and develop its working practices appropriately.

Methodology

The VTT's Human Factors team developed the Contextual Assessment of Organisational Culture (CAOC) methodology during the WOPS-project [18], [19], [21]. The methodology utilises two concepts, *organisational culture* and *core task*. The core task can be defined as the core demands and content of work that the organisation has to accomplish in order to be effective [12], [13]. The concept combines general goals of productivity, safety and employee well-being, which Vicente [25] has presented as criteria for effective sociotechnical systems. Thus, the core task concept can be used in assessing the central dimensions of the organisation's culture (see Figure 101). [21], [14], [19]

Organisational culture influences the definition of the core task, which in turn sets demands for the formation of culture. In CAOC-methodology, a conceptualisation of the target organisation's culture is made. The theoretical core task model constructed in the study acts as a point of comparison when examining the key features of culture.

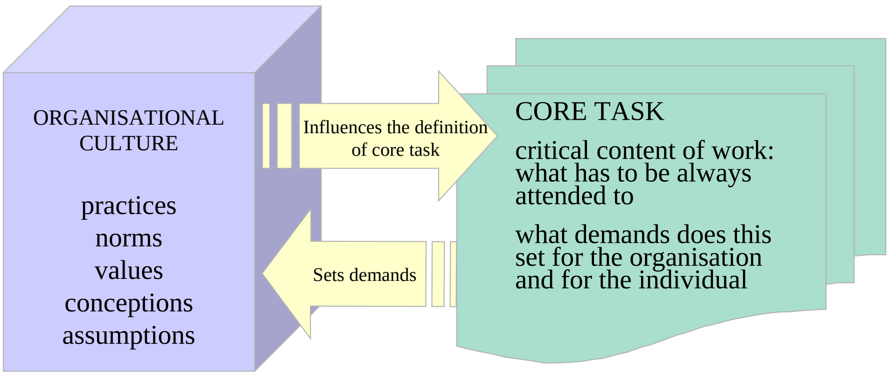

Figure 101. The central concepts of CAOC methodology.

In the WOPS -project, the CAOC-methodology was applied in studying regulatory work at Radiation and Nuclear Safety Authority of Finland's department of Nuclear Reactor Regulation [18], [19], [20] and in studying maintenance work at Loviisa NPP [14], [22]. The aim of these case studies was to model the core task of the organisation and characterise its organisational culture in order to assess how the culture supports perceiving and fulfilling the demands of the particular core task.

CAOC-methodology applied in regulatory work and maintenance work

Methods

The CAOC-methodology consists of both qualitative and quantitative methods. The methodology employs an iterative and multimethod research strategy based on method triangulation [3].

For core-task analysis an outline is needed of the characteristic features that influence work activity, the objectives of the work and the available resources. Core-task analyses have been performed by VTT in different contexts. The analytical method and the data acquisition methods have varied according to the subject area. In earlier studies in which the core-task concept has been utilised (see e.g. [11], [7], [12], [13]), the perspective has often been the nature of an individual's (e.g. nuclear power plant controller, anaesthetist) expertise and daily working practices. In cultural research the objective is the determination of the core task at the level of the whole organisation or its operational groups. The focus thus shifts away from the modelling of actual situations to the modelling of the general conditions of activity e.g. through group working and interviews. [21] In earlier studies, both of these have been done. In terms of development it is, however, important to examine working practices as communal phenomena. In regulatory study, the focus was on an inspector's core task, whereas maintenance study focused on the maintenance task in general.

Questionnaire method that was developed to be used in CAOC is called CULTURE, and it consists of four measuring instruments: measure of values, measure of individual conceptions, measure of the core task and measure of the psychological characteristics related to work. [22] The questionnaire was piloted in the case study at Radiation and Nuclear Safety Authority. At that time, the questionnaire did not contain a core task section. The core task section was added in maintenance culture study, and by its nature the section must always be carefully tailored to the given context. Measure of values and measure of individual conceptions utilises Cameron and Quinn's [2] model of competing values. The measure of psychological job characteristics is based on Hackman's et al. [4], [5], [6] Job Characteristics Model.

The qualitative methods of CAOC consist of interviews, observation of work activities and different workgroups or seminars [21]. Interviews serve various purposes in the cultural assessment process. Firstly, the interviewing of individuals working at different organisational levels and posts gives researchers an understanding of different job descriptions, language and concepts used in the organisation. Secondly, interviews provide an opportunity for material-oriented theming, which can be utilised e.g. in core task modelling. [21] Furthermore, interviews were used in tailoring the CULTURE-questionnaire. Interviews were also used in both case studies to interpret the statistical findings and statistical results, and to set hypotheses to be tested by statistical means.

Results from the regulatory work

On the basis of the core task analysis, we separated three roles of the regulator, the *expert role*, the *authority role* and the *public role*. The three roles set different demands for individual and organisational practices, which manifest as tensions in the regulatory culture (see Figure 102).

The requirements of all the three roles must be taken into account and balanced in action. The authority role requires perceiving safety relevant cues and practising mediated control to the power plants. In the authority role the inspector always makes judgements regarding the compliance with regulations, and he / she must also perceive things beyond the framework of the regulations. The public role requires reporting and informing openly to the public. Inspector has to make judgements about what information he / she is obliged to communicate and in what manner, and how this obligation is restricted by the demand for maintaining confidentiality. The expert role requires dialogue with other experts and self-criticism towards the level of own expertise. [19]

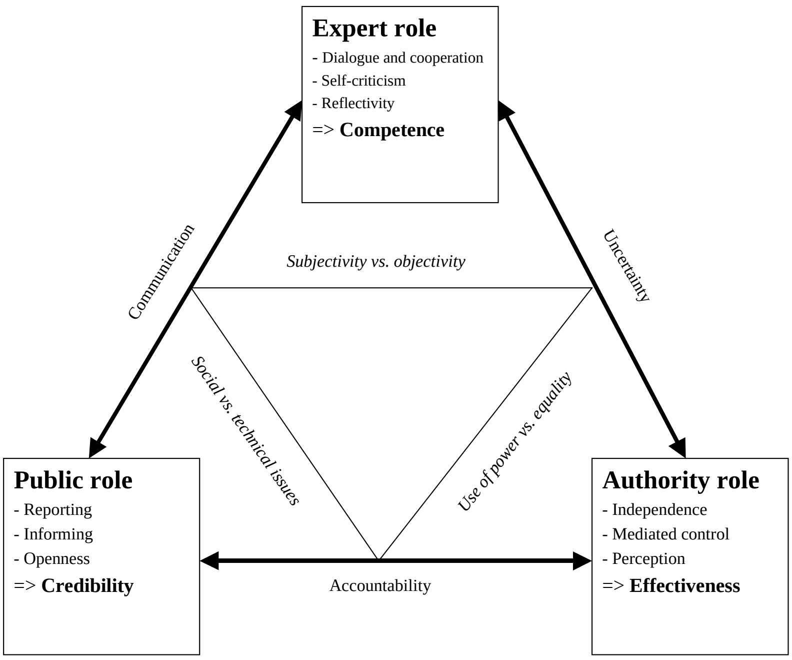

Figure 102. The conflicting demands of the regulatory work, adapted from Reiman and Norros [19].

Balancing between the roles is complicated by the inherent discrepancies between the roles (see Figure 102). Accountability sets a tension between the public role and the authority role, because the public role demands openness to the general public but the regulator has responsibilities also towards the power companies and cannot share all the information it has. The expert role differs from both other roles in subjectivity. Knowledge is always subjective, but the regulator speaks (to the public and to the power companies) with the voice of objective authority. The communication required in the public role is different from that required in the expert role. Uncertainty is allowed in the expert role (awareness of uncertainty is fundamental for the development of expertise) but not in the authority role. The authority role is the only role in which the use of power is allowed, and the public role demands taking into account social issues in addition to technical issues. [19]

Results from the maintenance work

Results of the maintenance culture interviews showed that despite the changes in the maintenance techniques and economical environment maintenance personnel do not see much change in their daily work. Goals of the maintenance are agreed upon and usually maintenance work and one's contribution to it is seen as meaningful and important to plant’s safe and reliable operation. There exists however some dissatisfaction among the personnel with the current organisational practices. The organisation was characterised as being bureaucratic and having centralised decision making. The current way of organising activities into technically specialised sections was experienced as complicating co-operation and co-ordination of work. It was pointed out that the organisation is able to reach its (safety) goals, but sometimes the current way of acting was seen as ineffective and unmotivating. Need for change was thus acknowledged, especially as the forthcoming change of generation sets challenges to recruiting and training of new employees.

Similar findings were obtained from the survey. Respondents perceived values related to change as lowest and safety values as highest in their section. Values related to financial efficiency were marked the second highest. This could be related to increased talk about optimisation of expenses and outsourcing during the recent years. Nevertheless the results imply that safety is perceived as the primary goal of maintenance and financial aspects more as internal requirements or constraints. [22], [14]

The model of maintenance core task was used in assessing how the culture of the maintenance department supported fulfilling the demands of the organisation's core task. This means that the different aspects of culture (values, climate, working practices) were considered from the perspective of how they supported or inhibited perceiving and fulfilling the demands of the maintenance core task. Due to maintenance work's complex, distributed and rule based nature achievement of a general view of plant's condition and flexible cooperation were the biggest challenges in the maintenance organisation. On the basis of interviews, we hypothesised that employees differ in the way they perceive their organisation and demands of their work. Different conceptions of right way of acting were assumed to complicate cooperation between sections. [14]

With the CULTURE-survey we identified different subcultures within the maintenance organisation. Age and length of service did create differences on change and development related factors. People with longer length of service at the company tended to view their organisation as more change oriented than the newer employees did. Recent changes (computerisation of work, economical pressures) were experienced as more stressful among older workers than among younger employees. Also hierarchical position in the company influenced the measured variables in a great degree. Mechanics had more negatively loaded meanings toward their organisation than the managers did. Previous studies in other contexts have also identified managers as having a tendency to perceive their organisation as more supportive than their subordinates [2], [18]. Differences were also identified between sections, some of which illustrate the nature of the work in question, whereas some reflect the cultural assumptions. For example electrical maintenance emphasised hierarchy values and occupational safety more than the instrumentation maintenance did, which reflects different requirements of work. In instrumentation maintenance, on the other hand, e.g. individual initiative and flexibility were endorsed more than in the other maintenance sections. [22]

It can be stated that the personnel at the maintenance department had several overlapping group identifications. Subcultures were also identified on the basis of personal orientation towards the maintenance core task [22], [23]. Different orientations towards the core task mean that the demands of work are constructed in different light. The meanings ascribed to maintenance and its demands have different content. The conception of the core task had a strong relationship to psychological job characteristics (for example meaningfulness of work) and perceptions of values. It seems that in order to develop working practices and culture it is essential to take into account that even the demands of the same task can be constructed differently.

The aim of the research was not only to assess the given culture, but also to give the personnel new concepts and new tools for reflecting on their organisation, their jobs and on appropriate working practices. The results of the study were presented to the personnel in different occasions and interpretations of the results were discussed together with the mechanics and foremen. Four working groups from different sections and hierarchical levels of the maintenance department were established in order to develop working practices on the basis of the results [14].

Conclusion

The CAOC-methodology was applied in two case studies. Although the methods used were much the same, the focus of the research differed. In the regulatory culture study the focus was on inspectors' work. Also, this study acted as a pilot project in developing and applying CAOC-methodology, and especially the CULTURE-questionnaire. Maintenance culture study focused on one particular maintenance organisation, without restricting itself to one job or task. Validation and development of the methodology continues, and the aim is to apply the methodology in other case studies and development projects in the future.

The CAOC-methodology contributes to the design and redesign of work in complex sociotechnical systems. The approach strives to enhance organisations' capability to assess their current working practices and the meanings attached to them and compare these to the actual demands of their basic mission and so change unadaptive practices. The need for organisational change and redesign of organisational structures should arise from careful analysis of the present state and the formulation of the preferred state on the basis of the (changed) requirements of the organisation's core task.

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14.3 Enhancing risk-informed ways of management of nuclear power plant fire situations (WOPS & METRI & FISRE)

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Abstract

The paper presents a method developed for the analysis of the management of NPP fire situations from the point of view of the control room. The way of analysis helps in gaining more understanding of fire situations as a context of operators' judgment and decision-making. The focus of the analysis is on the safety-critical significance of the way of forming an idea of the relevant matters of the situation and of the organizational support to making the assessments. The application of the method to particular fire situations makes it possible to identify the case-specific risk-informed ways of acting. The method offers a framework for the integration of the multidisciplinary expertise related to NPP fire situations. It can be used to support the management of fire situations and to improve the realism of fire risk analyses.

Introduction

The uncertainty, vagueness and complexity characteristics of NPP fire situations are known to be very demanding for the control room operators [3, 4]. This is reflected in the difficulties experienced in the development of procedures and training for fire situations.

The special features of the fire situations are problematic also in the assessment of human reliability. There are uncertainties in the modeling of the operator actions in probabilistic fire safety analyses [8]. The methodology of human reliability becomes more uncertain when the analyses to be applied have been adopted from other areas of PSA [1]. The way of applying the models concerning the internal events on the fire risk analyses is problematic because important information specific to fire situations may be lost [6]. This is especially obvious for the effects that a fire may have on the control room information and on the automatic devices and systems.

Because of these difficulties there is a need to develop a method which helps in gaining more understanding of the fire situations as the context of operators' judgment and decision-making. This kind of method could be used to support both the management of fire situations and the human reliability analyses concerning fires. In this study this challenge has been taken.

Goals

To enhance the safety of the management of fire situations two things are needed: deeper knowledge of the operators’ fundamental task demands and more co-operation among the relevant experts.

The main goal of the study has been to develop a method that helps 1) in defining the operators' task demands in the management of fire situations, 2) in identifying the developmental needs in the prerequisites of fulfilling these demands, provided by the plant organization and 3) in integrating the multidisciplinary expertise related to fire situations.

The knowledge gained by the method is intended to help in the improvement of the organizational support to the operators’ task performances and in the enhancement of the realism of fire risk analyses concerning human reliability.

Description of the method

The development of the method has been carried out as groups work in co-operation with different experts related to NPP fire situations. The integration of multidisciplinary expertise has been considered to be important because the different viewpoints help in gaining a more comprehensive picture of the special features of the fire situations.

The members of the group have been a simulator instructor, two control room shift supervisors and a development manager at Loviisa plant, an expert of the probabilistic fire safety assessment from the utility and three researchers from VTT, one of fire safety analysis, one of decision analysis and one of psychology.

The context and experience on local practices at Loviisa plant have been utilized in the development but the principles of the method, which are generic in nature, are also applicable at other nuclear power plants.

The origins of the approach lie in the psychological and decision analytic approach developed in VTT Industrial Systems [e.g. 5, 2, 7, 9].

The method is a way of conceptualizing, from the safety point of view, the effects of the context of fire situations on the demands concerning the operators' judgment and decision-making. The starting points of the analysis are the *main goal* and the *boundary conditions* of the management. The former stems from the ultimate need to avoid injuries and to maintain the nuclear safety of the plant process. The latter are the temporal, informational, operational and manpower resources. In the analysis the demands have been derived by taking into account the influence of the boundary conditions on the possibilities to reach the goals.

In order to reach the main goal the operators have to carry out certain *safety-critical control tasks* which have been defined in the procedures of the NPPs. The main categories of these tasks are:

- rescue operations
- control of the fire
- control of the nuclear process.

Taking care of the control tasks, for its part, requires the assessment of the current situation. An important prerequisite of a risk-informed way of acting is *to form an idea of the relevant matters* concerning rescue operations, fire fighting and process control. In the analysis the assessment has been divided into several subtasks which are called the *main assessment tasks*. The main categories of the identified tasks are:

- verification of the existence of the fire
- identification of the need for rescue operations
- assessment of the severity of the fire
- identification of the need for operations related to fire fighting and carried out by the control room staff
- identification of the need for shut-down, of the way of shut-down and of the safe state of the plant.

Another important prerequisite of a risk-informed way of acting is *the way of forming* an idea of the relevant matters, which depends on the *orientation* of the operators. In the analysis orientation has been considered as an attitude to carrying out the main assessment tasks and to the related co-operation with the fire fighting organisation and other relevant instances. In the former case orientation to the overall situation has been emphasized. As to the latter the focus of the analysis has been on the mediation of information among the different parts. In the definition of the *orientational criteria* for the way of acting three aspects of orientation have been adopted. They have been identified and defined by taking a systematic, analytic, subject-centred and problem-oriented view on the management of the situation. These aspects of orientation are:

- holistic
- ensuring
- critical.

The multidisciplinary expertise, enabled by the group work, has been utilized in the definition of the main assessment tasks and of the orientational criteria.

The analysis has been made on a general level but also a particular fire scenario has been designed in order to apply and test the analytic principles. The same scenario has been used as a basis of a simulator run in a fire and rescue drill, held in Loviisa plant in spring 2002. The experiences and feedback concerning the drill have been utilized in the development process.

Application of the method to particular fire situations

The application of the principles of the method to a particular fire scenario (see Table 12) makes it possible to define the case-specific risk-informed ways of acting. This is done by considering the operators' main assessment tasks with respect to the special features of the situation. Firstly, the specific features and difficulties related to the assessments and the risks inherent in the situation are identified and described. After that the ways of acting are defined by applying the orientational criteria to the specific task conditions. The way of analysis helps in considering the safety-critical significance of the alternative ways of acting in a particular situation.

The orientational criteria are preliminary and are supposed to become more specific as the result of the accumulation of the relevant knowledge concerning the essential aspects of the risk-informed way of acting.

Table 12. The framework for the case-specific analysis of the management of fire situations.

Main assessment tasks in management of fire situations	Case-specific difficulties and risks related to assessment of situation	Risk-informed way of acting		Organizational support to assessment of situation
		in assessment of situation	in co-operation	

The definition of the risk-informed ways of acting helps in identifying the possible deficiencies in the operators' prerequisites of making the relevant assessments. Here the following factors are of interest:

- division of labour in the control room
- procedures
- training
- prerequisites of communication
- information systems of the control room.

The recognition of the deficiencies makes it easier to identify the developmental needs concerning the organizational support to the management of fire situations.

Conclusions

The developed method offers a framework for the integration of the multidisciplinary expertise related to NPP fire situations. It can be used to support the management of the fire situations and to improve the realism of fire risk analyses.

The method helps in bringing up, from the safety point of view, the operators' most fundamental task demands in the management of NPP fire situations. The safety-critical significance of the way of forming an idea of the relevant matters of the situation emphasizes the importance of the organizational support to the operators' assessments.

The knowledge gained by the method can be used to improve the appropriateness of the *divison of labour* in the control room. It can also be utilized to enhance the context-sensitivity of the *procedures* and to identify the developmental needs related to them.

The method serves as a basis for the development of a tool which can be utilized in several ways in *simulator training* and in *fire and rescue drills*. It can be used in designing of the case-specific features of a particular fire scenario. The method provides a common frame of reference for integrating the expertise of simulator trainers, control room shift supervisors, fire risk analysts and representatives of fire fighting organization in the design. During the simulator run it can be used for observing the trainees' task performances and after the simulator run for giving feedback and for discussion. The cumulative scenario-specific knowledge can be utilized in identifying the developmental needs of training.

The method helps in enhancing both the trainers' and the trainees' understanding of the safety critical significance of the operators' ways of acting in the control of fire situations and in co-operation. It also offers a way to present the results of fire risk analyses in a concrete way to the operators. In addition, it helps to enhance the mutual awareness of the risk-informed ways of acting between the control room and the fire fighting organization.

In the development of the *fire risk analysis* the method can be used to support expert judgments and to facilitate the integration of the different experts' arguments. It helps in gaining more understanding of the effects of the special features of fire situations on the operators' action possibilities and of the sufficiency of the organizational support to the management of fire situations. The safety-critical significance of the prerequisites for forming an idea of the relevant matters concerning the situation could be emphasized by including these prerequisites in the definition of the safety requirements of the plant.

The integration of multidisciplinary expertise in the development of the method has contributed to positive feedback concerning the afore-mentioned fire and rescue drill held in Loviisa. The results of fire risk analyses were used in the design of the fire scenario that revealed facts unknown to the participating operators. This was considered to be very important from the operators' side.

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Title

FINNUS The Finnish Research Programme on Nuclear Power Plant Safety 1999–2002
Final Report

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 has concentrated on the themes of ageing, accidents and risks. The general objectives of the programme have been 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 have been prepared under the guidance of the Radiation and Nuclear Safety Authority (STUK), but the views of the Finnish power companies have been taken into consideration. Funding of the programme has been mainly from public sources. The annual volume of the programme has been about Euro 3.6 million and 30 person-years. The research has been co-ordinated and mainly conducted by the Technical Research Centre of Finland (VTT) with a significant contribution from Lappeenranta University of Technology (LTKK).

The effects of **ageing** on nuclear power plants have been studied intensively in order to evaluate the safe remaining lifetime of the components and the efficiency of the corrective measures. The programme has mainly concentrated on studies in ageing effects on material properties and degradation mechanisms of metallic structures, structural integrity and in-service inspection as well as monitoring methods including reinforced concrete structures as a new area. The **accident** theme has concentrated on operational aspects of nuclear power plant safety. The issues of nuclear fuel behaviour, reactor physics and dynamics modelling, thermal hydraulics and severe accidents were addressed under the theme by conducting both computational and experimental studies. In the **risk** field, attention has been paid to advanced risk analysis methods and their applicability, and to the evaluation of fire risks, safety critical applications of software-based technology, as well as human and organisational performance.

This final report summarises the goals and results of the programme. The programme has published 57 scientific articles, 233 mainly international conference papers, and 274 other reports. Six doctoral theses, two licentiate and 18 master's theses were completed. The total volume of the programme during the four years has been about 130 person-years and Euro14.4 million.

Keywords
FINNUS, nuclear power plants, reactor safety, corrosion, ageing, accidents, reactor physics, thermal hydraulics, modelling, fire safety, risk analysis, human factors

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