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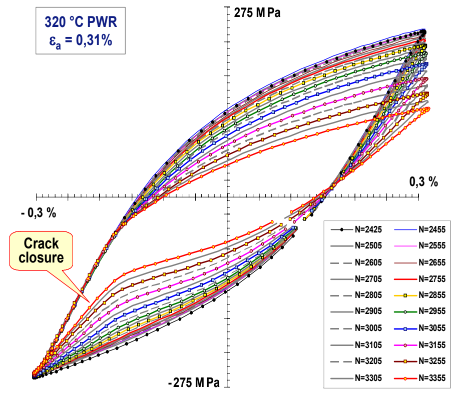

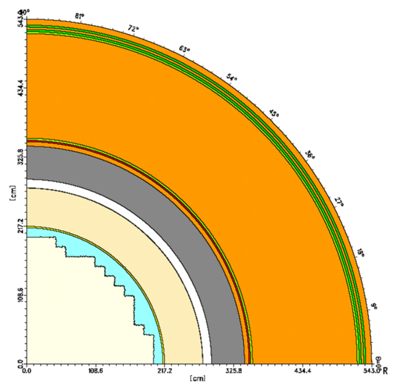

SAFIR

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

Interim Report

SAFIR

VTT TIEDOTTEITA – RESEARCH NOTES 2272

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Interim Report

Edited by

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

Abstract

SAFIR 2003–2006 is the present Finnish public research programme on nuclear power plant safety. The programme is administrated by the steering group that has been nominated by the Ministry of Trade and Industry (KTM). The steering group of SAFIR consists of representatives from Radiation and Nuclear Safety Authority (STUK), Ministry of Trade and Industry (KTM), Technical Research Centre of Finland (VTT), Teollisuuden Voima Oy (TVO), Fortum Oyj (Fortum), National Technology Agency of Finland (Tekes), Helsinki University of Technology and Lappeenranta University of Technology.

The six key research areas of SAFIR are 1) reactor fuel and core, 2) reactor circuit and structural safety, 3) containment and process safety functions, 4) automation, control room and IT, 5) organisations and safety management and 6) risk-informed safety management.

The research programme of the year 2004 involved 23 research projects, whose volume varied from a few person months to several person years. The total volume of the programme in 2004 was 4.9 million euros and 35 person years.

The research in the programme is performed primarily by the Technical Research Centre of Finland (VTT). Other research units responsible for the projects include Lappeenranta University of Technology, Fortum Nuclear Services Ltd and Helsinki University of Technology. In addition, there are a few minor subcontractors in some projects.

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

This report gives a summary of the results of the SAFIR programme for the period January 2003 – October 2004. During this period the programme produced 256 publications, four Doctoral, one Licentiate and six Master Thesis. The total volume of the programme in 2003–2004 was approximately 9 M€ and 67 person years.

Preface

SAFIR, The Finnish Research Programme on Nuclear Power Plant Safety 2003–2006 is the newest link in the chain of Finnish national research programmes in nuclear area. Organisation of public nuclear energy research as national research programmes was started in 1989 by the Ministry of Trade and Industry (KTM). Since then national programmes have been carried out in the fields of operational aspects of safety (YKÄ 1990–1994, RETU 1995–1998), structural safety (RATU 1990–1994, RATU2 1995–1998), and in FINNUS 1999–2002 that combined the operational aspects and structural safety. Simultaneously the research was carried out in nuclear waste management programmes (JYT 1989–1993, JYT2 1994–1996, JYT2001 1997–2001).

In parallel with the public programmes research has been carried out in the Finnish Fusion Research Programme (FFUSION2) 1993–2002, programmes on Advanced Light Water Reactor concepts (ALWR) 1998–2003 and a project on component life management 1999–2003, partly funded by the National Technology Agency (Tekes). Currently fusion research continues in the FUSION (2003–2006) and nuclear waste management research in the KYT (2002–2005) programme.

The steering group of SAFIR consists of representatives from Radiation and Nuclear Safety Authority (STUK), Ministry of Trade and Industry (KTM), Technical Research Centre of Finland (VTT), Teollisuuden Voima Oy (TVO), Fortum Oyj (Fortum), National Technology Agency of Finland (Tekes), Helsinki University of Technology and Lappeenranta University of Technology.

In 2003 the main funding sources of the programme were KTM, STUK, VTT, TVO and Fortum. At the beginning of 2004 there was a major change in the funding structure of the programme due to a change in the Finnish legislation on nuclear energy. The funding by KTM, STUK, TVO and Fortum was replaced by funding from a separate fund of the State Nuclear Waste Management Fund (VYR). This VYR-funding is collected from the Finnish utilities Fortum and TVO with respect of their MWth shares in Finnish nuclear power plants.

The main funding sources of the programme in 2004 were the State Nuclear Waste Management Fund (VYR) with 2.7 M€ and Technical Research Centre of Finland (VTT) with 1.3 M€. The rest of the funding originated from several partners.

The six key research areas of SAFIR are 1) reactor fuel and core, 2) reactor circuit and structural safety, 3) containment and process safety functions, 4) automation, control room and IT, 5) organisations and safety management and 6) risk-informed safety management.

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

This report has been prepared by the programme management in cooperation with the project leaders and project staff.

Contents

1. Introduction

Nuclear safety research in Finland consists of three components: regulatory research, utility research and public research. Regulatory research and utility research whose total annual volume exceeds the volume of public research is strictly separated from the public research programme.

Public nuclear safety research provides the necessary conditions for retaining the knowledge needed for ensuring the continuance of safe and economic use of nuclear power, for development of new know-how and for participation in international cooperation. The role of the public safety research is illustrated in Figure 1.

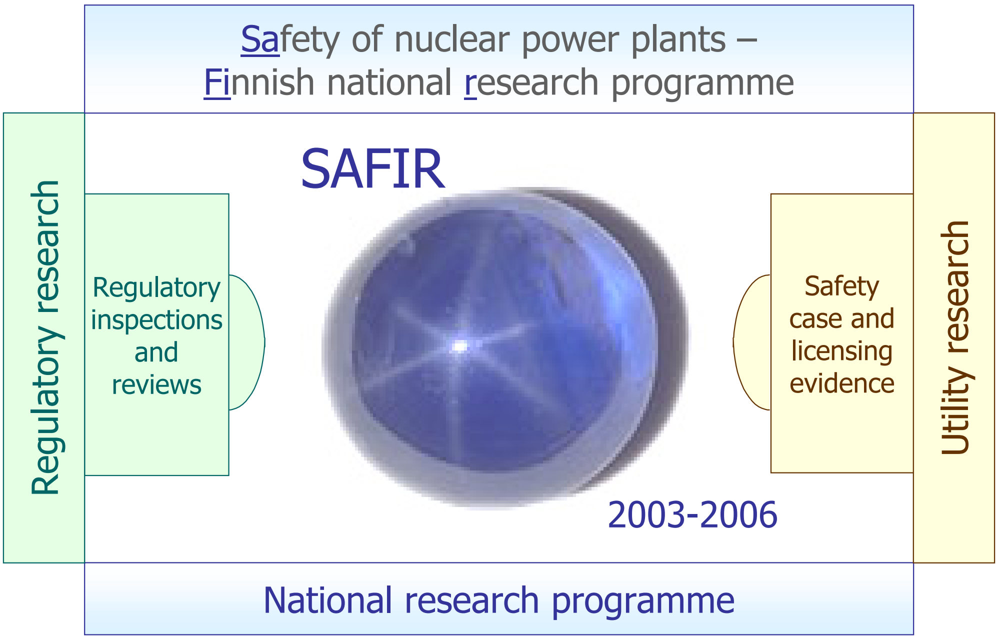

Figure 1. The roles of public, regulatory and utility research in the Finnish research field.

Planning of the new research programme was carried out by a special ad hoc committee. The work of the committee has been presented in the framework plan [1]. The plan has been made for the period 2003–2006, but it is based on safety challenges identified for a longer time span as well. It was recognized that the safety challenges set by the existing plants and the new plant unit, as well as the ensuing research needs do converge to a great extent.

The framework plan [1] defines the important research needs related to the safety challenges, such as the ageing of the existing plants, technical reforms in the various areas of technology and organisational changes. The research into these needs is the programme's main techno-scientific task. In addition, the programme has to ensure the maintenance of know-how in those areas where no significant changes occur but in which dynamic research activities are the absolute precondition for safe use of nuclear power. The current research programme take advantage of the results obtained and lessons learned in the former national research programme FINNUS [2, 3], as well as all the preceeding programmes since 1989.

SAFIR 2003–2006 is the present Finnish public research programme on nuclear power plant safety. The programme is administrated by the steering group that has been nominated by the Ministry of Trade and Industry (KTM). The steering group of SAFIR consists of representatives from Radiation and Nuclear Safety Authority (STUK), Ministry of Trade and Industry (KTM), Technical Research Centre of Finland (VTT), Teollisuuden Voima Oy (TVO), Fortum Oyj (Fortum), National Technology Agency of Finland (Tekes), Helsinki University of Technology and Lappeenranta University of Technology. The major partners of SAFIR are shown in Figure 2.

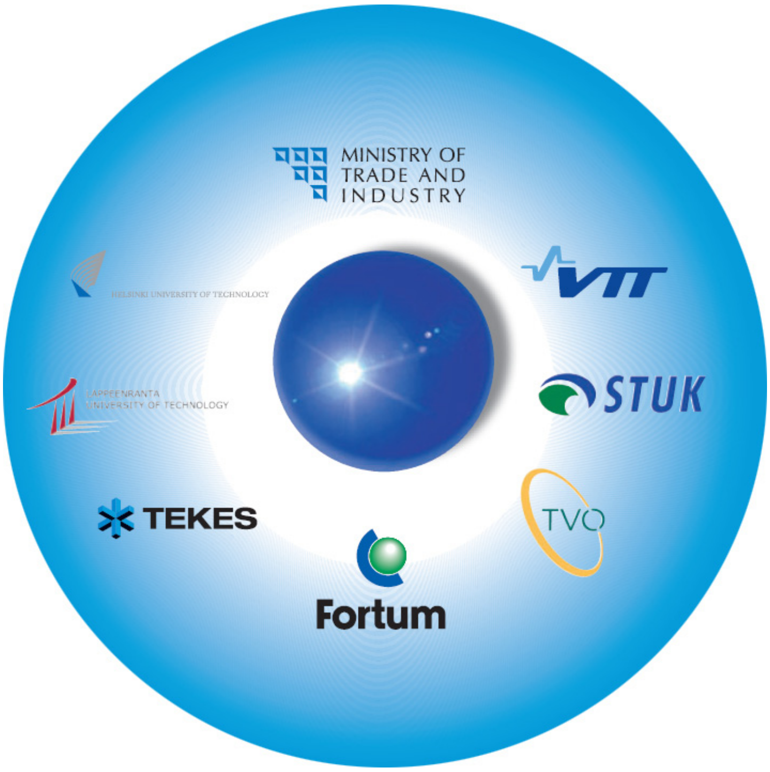

Figure 2. All the key players of the Finnish nuclear field are represented in the SAFIR steering group.

The six key research areas of SAFIR are 1) reactor fuel and core, 2) reactor circuit and structural safety, 3) containment and process safety functions, 4) automation, control room and IT, 5) organisations and safety management and 6) risk-informed safety management.

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

One of the main goals of the SAFIR programme is the education of the new generation. In this most ‘nuclear-specific’ research area of SAFIR, this task is particularly pronounced. All SAFIR research areas have links both between the various projects in the area and to neighbouring research areas. The most vital connections of the reactor fuel and core area are with the reactor circuit and structural safety area and with the containment and process safety functions area.

In 2004 there have been two research projects in reactor fuel and core area, the Enhanced methods for reactor analysis (EMERALD) project dealing with reactor physics and dynamics and the High-burnup upgrades in fuel behaviour modelling (KORU) project dealing with the fuel research. Both of these projects continue from the year 2003.

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

In 2004 there were four research projects in this area. The Integrity and life time of reactor circuits (INTELI) project is a very large one. The main objective is to assure the structural integrity of the main components of the reactor circuit of the nuclear power plant and to study the typical ageing mechanisms affecting the integrity of main components during the life-time of the reactor. The main components included in the scope of the project are reactor pressure vessel with nozzles and internals, piping of reactor circuit and other components (steam generators, pumps, valves, pressurizer, heat-exchangers). Oxide modelling is studied in the LWR oxide model for improved understanding of activity build-up and corrosion phenomena (LWROXI) project and the containment is studied in two separate projects, Ageing of the function of the containment building (AGCONT) project and Concrete technological studies related to the construction, inspection and reparation of the nuclear power plant structures (CONTECH) project. The INTELI, AGCONT and CONTECH project continue from the year 2003 with the small CONAGE project of 2003 being connected to the AGCONT project.

Containment and process safety functions area covers simulation of nuclear power plant processes, calculational thermal hydraulics and multiphysics approaches using several codes, experimental thermal hydraulics at Lappeenranta University of Technology (LUT) and various severe accidents projects, where both experimental and theoretical work is included.

Multiphysical approaches, strong coupling of experimental and theoretical work and active follow-up and participation in international research programmes are characteristic to the projects in this research area. In this field, with several very ‘nuclear-specific’ projects, fostering of a new generation of experts has a vital role, too.

This by far the largest research area of SAFIR hosted in 2004 altogether 11 research projects. They were the Wall response to soft impact (WARSI), Impact Tests (IMPACT), The integration of thermal-hydraulics (CFD) and finite element (FEM) computer codes in liquid and solid mechanics (MULTIPHYSICS), APROS modelling of containment pressure suppression systems (TIFANY), Thermal hydraulic analysis of nuclear reactors (THEA), Severe accidents and nuclear containment integrity (SANCY), Fission product gas and aerosol particle control (FIKSU), Development of aerosol models for NPP applications (AMY), Archiving experiment data (KOETAR), Condensation pool experiments (POOLEX) and PACTEL OECD project planning (PACO). The WARSI, TIFANY, THEA, SANCY, FIKSU, AMY, KOETAR and POOLEX projects continue from the year 2003. In 2003 the area included also a relative small OTUS project on emergency preparedness planning.

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

The focus of research in **Automation, control room and IT** is on the new technologies that are emerging at nuclear power plants both via new plants and via renewal of automation and control rooms in the existing plants.

Two research projects were included in 2004: Interaction approach to development of control rooms (IDEC) project aimed at formulating a scientifically founded method for the evaluation of human-system interfaces of complex industrial systems and a pre-project study on Influence of RoHS -directive to reliability of electronics (ROVEL) was performed. The IDEC project continued from the year 2003. In 2003 the area included also the APSREM project on requirements management. Additionally, the research area is an important forum of information exchange on the work done beyond SAFIR in Finland and in some international projects.

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

In 2004 the work in SAFIR in this area was performed in the project Organisational culture and management of change (CULMA) and pre-project Disseminating tacit knowledge in organisations (TIMANTTI). The CULMA project continued from the year 2003. As well as in other research areas, the projects involved also participation in international research projects and working groups.

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

The research area included in 2004 two projects: Potential of Fire Spread (POTFIS) project, where the goal is to develop deterministic and stochastic sub-models to the same level as other branches of PSA. The major strategic problem in SAFIR is the ability to predict potential of fire spread in given scenarios. The Principles and Practices of Risk-Informed Safety Management (PPRISMA) project deals with the whole scope of risk-informed methods and application areas related to safety of nuclear power plants. Both projects continued from the year 2003.

The research performed in the projects is supervised by six reference groups. The programme is managed by the coordination unit VTT Processes, the programme director, the project co-ordinator and the project managers of the individual research projects.

There were 20 research projects and the administration project going on during the year 2003 in SAFIR. The AMY-project joined in the programme after the publication of the annual plan [4]. The projects and their division into the six research areas have been illustrated in Table 1. The extent of the projects varied from a few person months into several person years. Most of the projects have been planned to continue throughout the entire four-year span of the SAFIR programme. Four of the projects were under the 'umbrella' of SAFIR without KTM-funding. CONTECH was funded by STUK and several other partners, OTUS by STUK and VTT, and TIFANY and AMY by Tekes and the utilities.

Table 1. The research projects of SAFIR in 2003.

Group	Project and principal research organisation	Acronym	Funding k€		Volume person years	
			plan	realised	plan	realised
1.						
	Enhanced methods for reactor analysis *VTT Processes*	EMERALD	525	487	4,48	4,36
	High-burnup upgrades in fuel behaviour modelling *VTT Processes*	KORU	210	220	1,95	2,11
2.						
	Integrity and life time of reactor circuits *VTT Industrial Systems*	INTELI	1057	1019	7,5	7
	Ageing of the function of the containment building *VTT Building and Transport*	AGCONT	13	13	0,1	0,1
	Participation in the OECD NEA task group concrete ageing *VTT Building and Transport*	CONAGE	9,48	9,5	0,07	0,07
	Concrete technological studies related to the construction, inspection and reparation of the nuclear power plant structures *VTT Building and Transport*	CONTECH	100,5	100,5	0,7	0,7
3.						
	Wall response to soft impact *VTT Industrial Systems*	WARSI	137,7	138,4	1,2	1,2
	The integral code for design basis accident analyses *Fortum Nuclear Services*	TIFANY	206,4	203,3	1,55	1,53
	Thermal hydraulic analysis of nuclear reactors *VTT Processes*	THEA	187	165,8	1	1
	Severe accidents and nuclear containment integrity *VTT Processes*	SANCY	315,6	314	1,62	1,56
	Fission product gas and aerosol particle control *VTT Processes*	FIKSU	102	112	1	1
	Development of aerosol models for nuclear applications *Fortum Nuclear Services*	AMY	191	150	1,4	1,9
	Emergency preparedness supporting studies *VTT Processes*	OTUS	50	50	0,43	0,41
	Archiving experiment data *Lappeenranta University of Technology*	KOETAR	60	60	0,5	0,6
	Condensation pool experiments *Lappeenranta University of Technology*	POOLEX	152,5	163,9	1,4	2

4.						
	Interaction approach to development of control rooms *VTT Industrial Systems*	IDEC	140	140	1,1	1,1
	Application possibilities of systematic requirements management in the improvement of nuclear safety in Finland *RAMSE Consulting*	APSREM	50	50	0,28	0,44
5.						
	Organisational culture and management of change *VTT Industrial Systems*	CULMA	205,7	201	1,5	1,45
6.						
	Potential of fire spread *VTT Building and Transport*	POTFIS	158	158,5	0,9	0,7
	Principles and practices of risk-informed safety management *VTT Industrial Systems*	PPRISMA	258,55	245	2,08	2,07
0.	SAFIR Administration and information (2002-2003) *VTT Processes*	SAHA	116,7	107,9	0,86	0,81
	Total		**4246,15**	**4108,76**	**31,62**	**32,11**

SAFIR research programme consisted at the end of the year 2004 of 23 research projects in the six research areas. The planned volume of the programme was 35 person years and 4.9 million €. The volume of the projects varied from some person months up to several person years. Most of the projects have been planned to continue throughout the entire four-year span of the SAFIR programme [4]. The projects and their division into the six research areas have been illustrated in Table 2.

Distribution of funding in the SAFIR research areas in 2003–2004 is shown in Figure 3, and distribution of person years in Figure 4, respectively. It can be seen that the total volume of the programme increased in 2004 from the previous year by some 0,8 million euros and 2,4 person years. The increase was directed into the research areas 1–3. Distribution of funding and person years in the six research areas of SAFIR have been illustrated in Figures 5 and 6, respectively.

Table 2. The research projects of SAFIR in 2004 (funding and volumes according to updated plans).

Group	Project	Acronym	Funding k€	Volume person years
1.				
	Enhanced methods for reactor analysis	EMERALD	570	4,81
	High-burnup upgrades in fuel behaviour modelling	KORU	281	2,57
2.				
	Integrity and life time of reactor circuits	INTELI	1082	6,57
	LWR oxide model for improved understanding of activity build-up and corrosion phenomena	LWROXI	86	0,62
	Ageing of the function of the containment building	AGCONT	34	0,24
	Concrete technological studies related to the construction, inspection and reparation of the nuclear power plant structures	CONTECH	107,5	0,85
3.				
	Wall response to soft impact	WARSI	140,5	1,14
	Impact tests	IMPACT	202	0,95
	The integration of thermal hydraulics (CFD) and finite element (FEM) computer codes in liquid and solid mechanics	MULTIPHYSICS	116	0,69
	APROS modelling of containment pressure suppression systems	TIFANY	192,4	1,38
	Thermal hydraulic analysis of nuclear reactors	THEA	158	0,95
	Severe accidents and nuclear containment integrity	SANCY	305	1,19
	Fission product gas and aerosol particle control	FIKSU	55,5	0,43
	Development of aerosol models for NPP applications	AMY	212	1,90
	Archiving experiment data	KOETAR	60	0,57
	Condensation pool experiments	POOLEX	256	2,24
	PACTEL OECD project planning	PACO	60	0,19
4.				
	Interaction approach to development of control rooms	IDEC	196	1,34
	Influence of RoHS-directive to reliability of electronics – preproject	ROVEL	20	1,11
5.				
	Organisational culture and management of change	CULMA	210	1,52
	Disseminating tacit knowledge in organisations – preproject	TIMANTTI	27	0,43
6.				
	Potential of fire spread	POTFIS	158	0,95
	Principles and practices of risk-informed safety management	PPRISMA	248,5	2,05
0.	SAFIR Administration and information	SAHA	130,16	0,81
	Total		**4907,56**	**34,71**

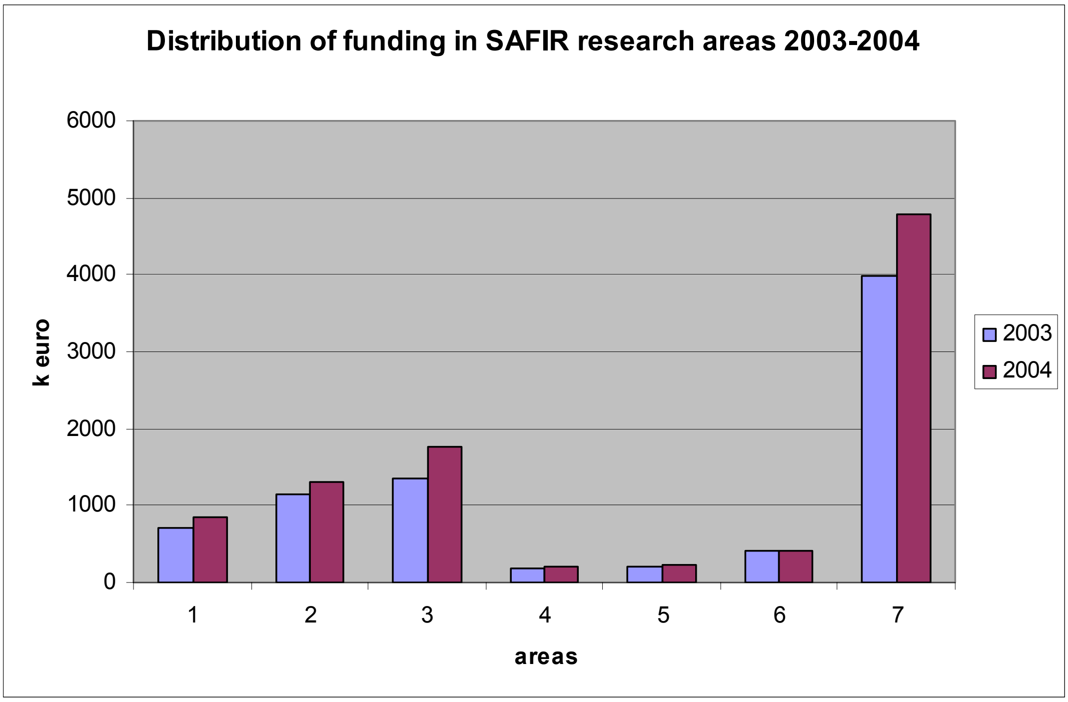

Figure 3. Distribution of funding in the SAFIR research areas in 2003–2004.

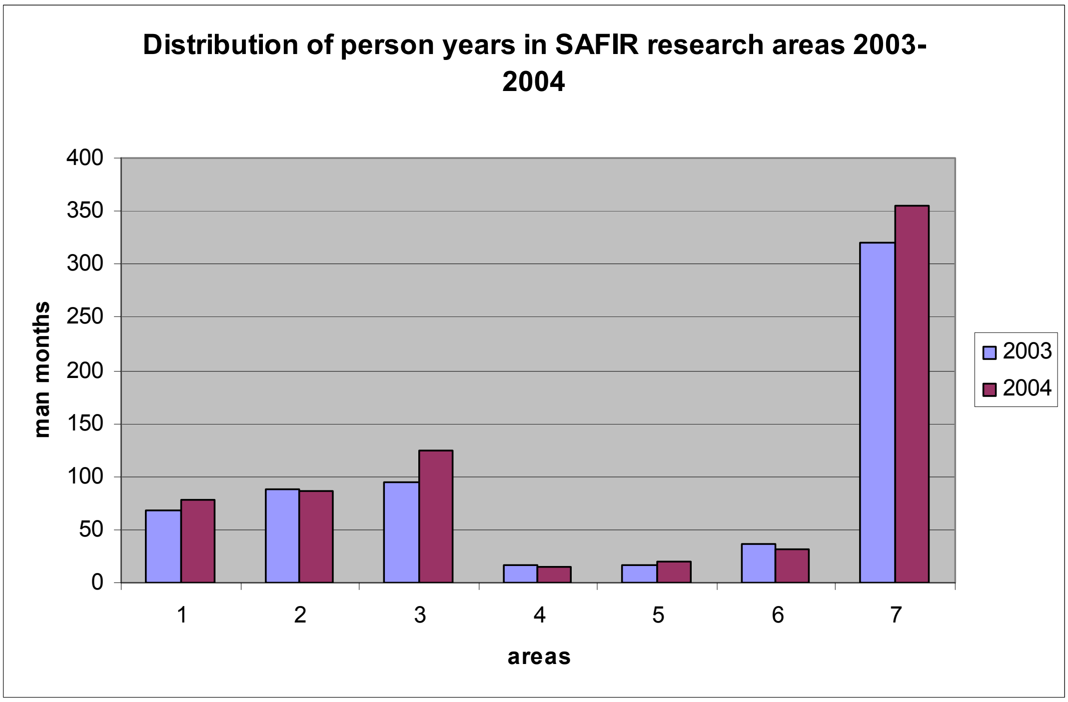

Figure 4. Distribution of person years in the SAFIR research areas in 2003–2004.

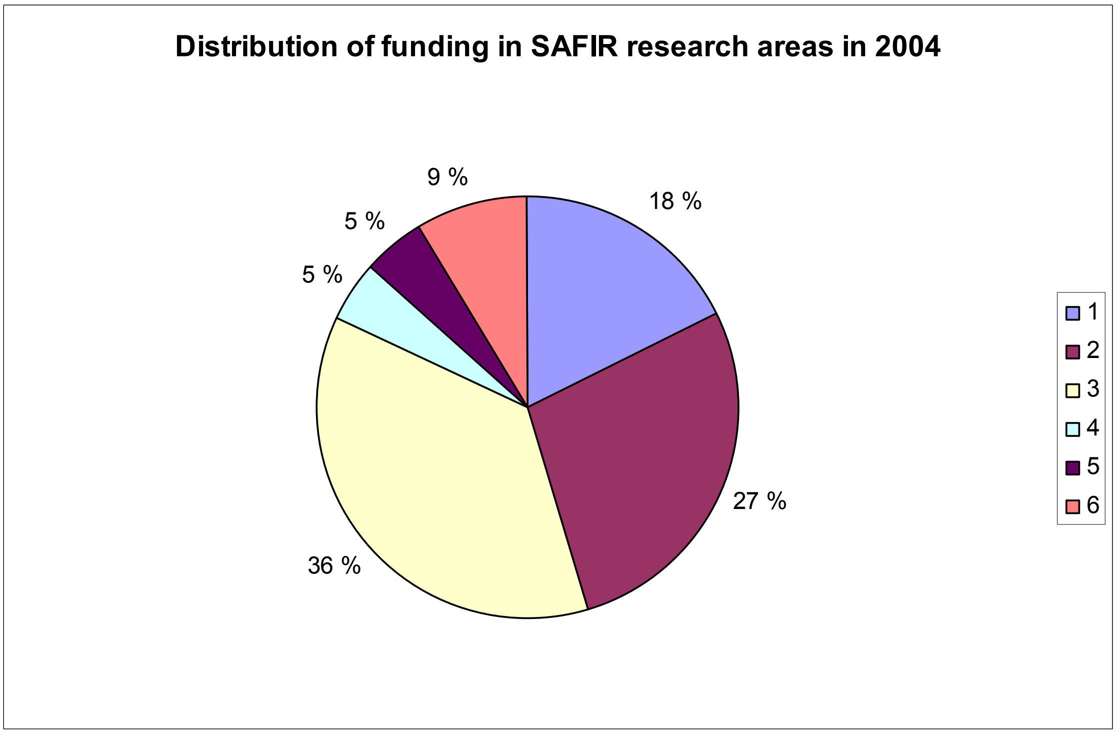

Figure 5. Distribution of funding in the SAFIR research areas in 2004.

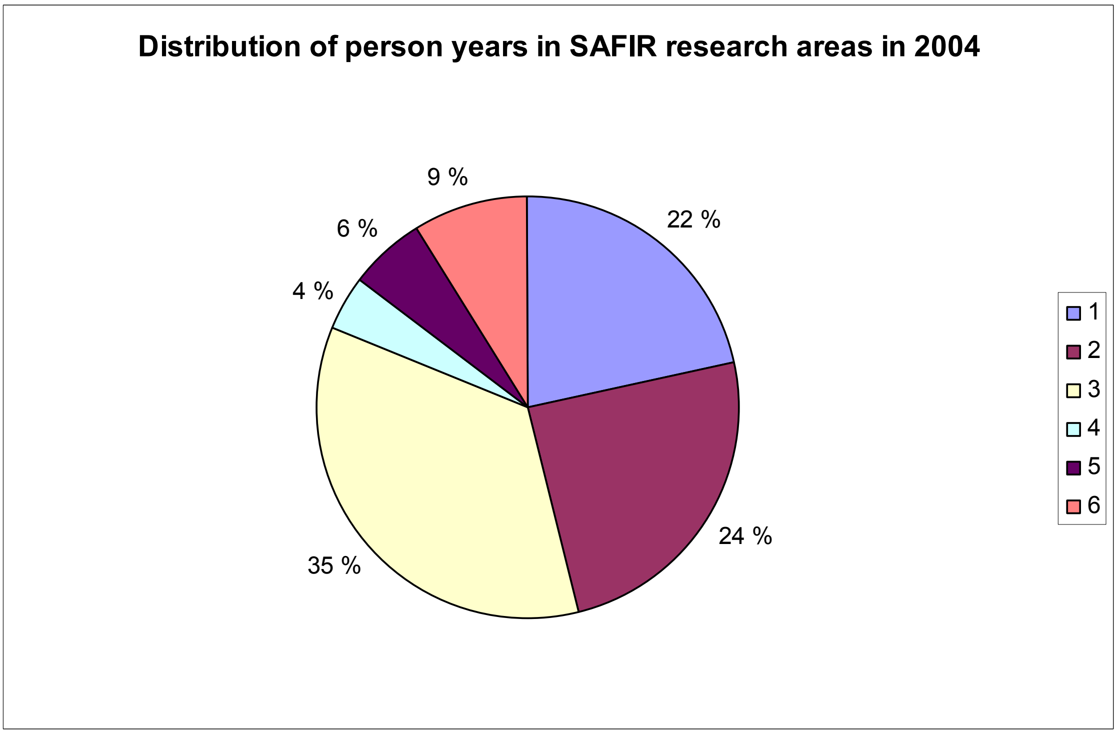

Figure 6. Distribution of person years in the SAFIR research areas in 2004.

The programme has produced 256 publications in 2003–2004. Major part of the publications consisted of conference papers and extensive research institute reports. The number of scientific publications as well as the total number of publications varied greatly between the projects, as indicated in Table 3. The average number of publications is 3.8 per person year, and the average number of scientific publications is 0,34 per person year. The number of scientific publications is low due to the fact that part of the projects have deliberately aimed at publication of the results as research institute reports.

Table 3. Publications in the SAFIR projects in 2003–2004.

Project	Scientific	Conference papers	Res. inst. reports	Others	Total	Volume pers. year
EMERALD	2	20	8	3	33	9,17
KORU	0	2	1	1	4	4,68
INTELI	8	16	12	19	55	13,57
LWROXI	0	1	1	0	2	0,62
AGCONT	0	0	1	0	1	0,34
CONAGE	*0*	*0*	*0*	*0*	*0*	*0,07*
CONTECH	0	0	4	0	4	1,55
WARSI	0	0	2	1	3	2,34
IMPACT	0	0	0	0	0	0,95
MULTIP	0	0	3	3	6	0,69
TIFANY	0	0	13	0	13	2,91
THEA	0	3	1	0	4	1,95
SANCY	1	0	8	1	10	2,85
FIKSU	0	2	5	0	7	1,43
AMY	0	2	4	2	8	3,8
OTUS	*0*	*0*	*2*	*0*	*2*	*0,41*
KOETAR	0	0	2	0	2	1,17
POOLEX	0	0	2	5	7	4,24
PACO	0	0	0	0	0	0,19
IDEC	2	7	4	0	13	2,44
ROVEL	0	0	0	0	0	0,11
APSREM	*0*	*0*	*1*	*0*	*1*	*0,44*
CULMA	2	5	3	5	15	2,97
TIMANTTI	0	3	0	0	3	0,43
POTFIS	7	9	1	12	29	1,65
PPRISMA	1	6	15	8	30	4,12
SAHA	0	0	3	1	4	1,66
Total	**23**	**76**	**96**	**61**	**256**	**66,8**

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

Table 4. Academic degrees awarded in the projects.

Project	Doctor (DTech, PhD)	Licentiate (LicTech, LicPhil)	Master (MScTech, MSc, MA)	Total
EMERALD	-	1	1	2
INTELI	2	-	-	2
AMY	-	-	1	1
POOLEX	-	-	2	2
IDEC	-	-	1	1
POTFIS	1			1
PPRISMA	1	-	1	2
Total	**4**	**1**	**6**	**11**

The programme management bodies, the steering group and the six reference groups, have met on regular basis 3–4 times annually. The ad hoc groups that have a vital role in some areas with many projects have carried out successfully their tasks. The ad hoc groups have met upon the needs of the specific project. All these groups will be regularly informed using standard progress reports. Figure 7 illustrates the structure of the SAFIR programme with the research projects forming the hot red core of the programme, the six reference groups and the various ad hoc groups having the principal responsibility of scientific guidance and surveillance of the various research projects, as depicted with the yellow layer encircling the red core. The steering group, depicted as the blue layer, administrates the entire research programme thus keeping the SAFIR 'jewel' together.

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

The information on the research performed in SAFIR has been communicated formally via the quarterly progress reports, the annual plans [4, 6] and annual reports [5] of the programme and the www-pages of the programme. Additional information has been given in seminars organised by various research projects. The detailed scientific results have been published as articles in scientific journals, conference papers, and separate reports. During the year 2004 a brochure on the SAFIR programme was published, too.

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

The report contains presentation of the main scientific achievements of the projects in Chapters 2–24 both in the format of project overviews and special technical reports. The Appendices give further statistical information on the programme. Appendix A contains the publications of the projects, Appendix B lists the international co-operation connections, Appendix C contains the list of academic degrees awarded and Appendix D list the members of the steering group, the reference groups and the scientific staff of the projects.

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2. Enhanced methods for reactor analysis (EMERALD)

2.1 EMERALD summary report

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Abstract

The purpose of the EMERALD project is to achieve a reliable computer code system for all necessary steady-state and transient reactor analyses. Development of models and codes as well as validation of the latter have included neutron cross section studies, nodal, Monte Carlo and deterministic transport methods, criticality safety calculations and the improvement of the dynamics codes to allow coolant flow reversal. Several international meetings have been arranged and the results of the codes have been tested through benchmark calculations and other comparisons.

Introduction

VTT Processes has created a comprehensive computer code system and striven to maintain competence for carrying out all reactor physics calculations needed in Finland including the necessary safety analyses. New fuel and reactor designs, new loading strategies, and the continuing trend towards higher fuel burnups make it necessary to further improve as well as validate the code system to be able to cope with the new challenges.

Main objectives

The main objective is to accomplish a unified, complete, up-to-date, easy-to-use, and flexible entirety consisting of both programs acquired from elsewhere and programs that are the result of own development. The code system should cover the whole range of calculations, from handling of basic nuclear data, i.e. cross section libraries, over fuel and core analyses in normal operating conditions to transient and accident studies using coherent models and methods. It should be possible to follow the whole life cycle of the nuclear fuel from a reactor physics point of view until its final disposal. The same or similar models can often be used in both the static and the dynamic calculations.

Additionally, it is of special importance in today´s situation, when the use of nuclear power is increased at the same time as the present generation of nuclear experts are gradually retiring from work, to maintain competence and train new personnel. Co-operation with the technical and other universities is necessary to make new students interested in this branch of science and thus ensure that the nuclear plants in Finland will be in the hands of competent people in the future, too. The tasks of the project have provided and will continue to provide excellent possibilities for university students to perform work for their academic degrees.

Reactor physics

The work on reactor physics has been divided into four subprojects, which deal with nuclear cross sections, nodal methods, transport methods, and criticality safety & isotope concentrations.

The results obtained in any reactor analysis is very much dependent on the quality of the cross section data that describe the interaction of the neutrons with the surrounding medium. Cross sections are often modified for different programs and different problems, but nevertheless originally based upon one of the existing basic cross section libraries. Comparisons between recent versions of the American ENDF/B, NEA Data Bank´s (originally European) JEF/JEFF and the Japanese JENDL libraries have been performed [5, 6]. These studies are reported in a separate article below.

A new BWR core simulator code was developed at VTT Processes during SAFIR´s predecessor FINNUS. Its name ARES was derived from the words **A**FEN **Re**actor **S**imulator, where AFEN stands for the neutronics solution method used in the program, which is based upon the Analytic Function Expansion Nodal model [8]. The simulator has been further improved and tested during the initial phase of the EMERALD project and has now also been used for EPR calculations. ARES is useful when reference calculations for foreign commercial codes are needed, i.e. for finding and evaluating problem areas and safety margins. It can produce burnup distributions for transient calculations in independent safety studies and also makes it possible to test new models and ideas in reactor core analysis.

With the continuous development of more efficient computers, it has become possible to use highly accurate, but also very time-consuming Monte Carlo methods in many reactor physics problems. At VTT Processes, the MCNP code has been frequently utilized.

A deterministic 3D radiation transport code MultiTrans has been created at VTT. It is based upon the so-called tree multigrid technique for improving the mesh used in the calculation for solving a certain problem. It has been used for BNCT dose planning and photon-electron dose calculations in more traditional radiotherapy. In benchmark studies, MultiTrans has now also been successfully applied to reactor fuel and core criticality calculations [4].

Except for the main reactor analysis codes themselves, there are many auxiliary codes necessary to be able to use the main codes efficiently. Several such codes have been installed and utilized during the EMERALD project, e.g. a new NJOY version and JANIS for the manipulation of cross section data, and BOT3P for neutron transport code input and output treatment (Figure 8).

Nuclear criticality safety is a term meaning all actions aimed at the prevention of a criticality accident. Originally, fresh fuel was assumed in the criticality safety analyses for the storage and transport of spent fuel assemblies also. Nowadays, the increased use of burnable absorbers and the trend towards higher enrichments and burnups call for a more realistic approach that pays attention to the changes in assembly composition during irradiation. Taking the burnup related changes in isotopic concentrations into account is referred to as burnup credit (BUC), the use of which can reduce the conservatism in the design of spent fuel storage and transport equipment, thus leading to considerable savings in the costs.

A number of comparisons with experimental results, as well as results obtained by calculation elsewhere, have been made using the CASMO-4 and Monteburns 1.0 codes [10]. CASMO, Studsvik Scandpower´s code for generation of cross section data, is widely used all over the world for ICFM studies on both BWR and PWR reactors as well as for other applications.

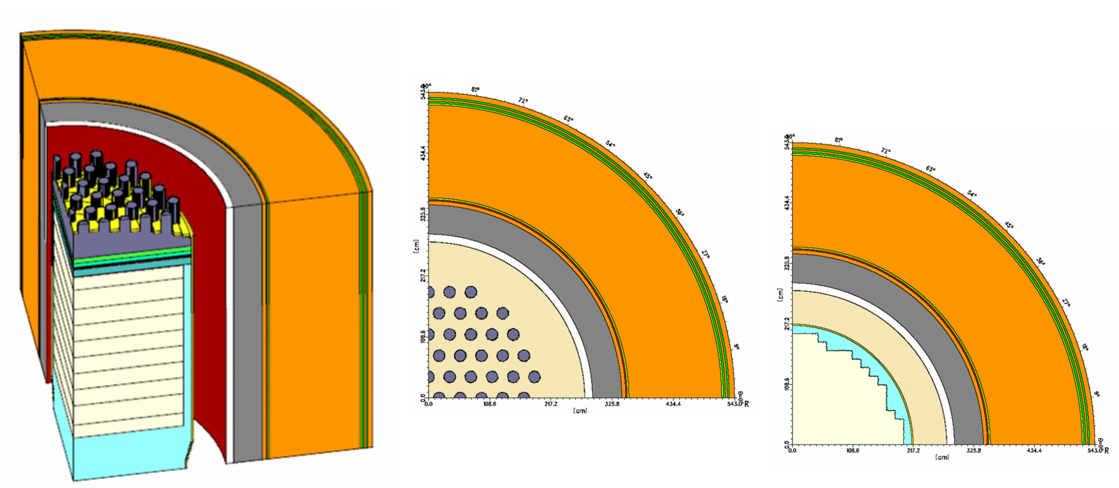

Figure 8. Modelling of the Olkiluoto BWR core by BOT3P for transport calculations using the TORT code.

Reactor dynamics

With the reactor dynamics codes developed at VTT, i.e. TRAB, TRAB-3D, and HEXTRAN, it is possible to perform transient analyses for cores with square or hexagonal fuel bundle geometry for boiling water as well as pressurized water reactors. An important limitation of the present codes, however, has been their inability to handle coolant flow reversal in a flow channel, a phenomenon that can be encountered in e.g. BWR ATWS cases or at natural circulation conditions in connection with VVER power excursions.

As a fast solution to this problem, two existing computer codes, the TRAB-3D reactor dynamics code and the SMABRE thermal hydraulics code [2, 7], have been coupled together using an internal coupling scheme. This means that TRAB-3D performs only the neutronics and heat transfer calculation, whereas SMABRE takes care of the hydraulics calculation of the whole cooling circuit including the reactor core. The solution will be somewhat less accurate than with the standard TRAB-3D version, but the coupled code TRAB-3D/SMABRE should be able to calculate transients with flow reversal in the reactor core or the core by-pass with reasonable accuracy. Also, the new internal coupling will make the future modelling of in-core cross-flows in PWR (such as EPR) open core geometry possible.

The coupling of TRAB-3D and SMABRE is described in more detail in a special article below.

As a future solution to various reactor dynamics problems, the highly accurate PLIM thermal hydraulics solver has been developed during earlier research programs already. It will for instance be able to track moving fronts, like boron fronts, accurately and also to deal with flow reversals. It has previously been successfully applied to several limited problems, but its use in full BWR circuit dynamics calculations, where the hydraulics solution is strongly coupled to other phenomena, i.e. neutronics and heat transfer, has proved difficult and sensitive to small disturbances.

In order to make the solver more robust, and to enhance its modelling capabilities, a new basic solution method has been created. The theoretical work on this model has been completed during the first phase of the EMERALD project. The work will continue with a reprogramming of the CFDPLIM computer code. The new solver combines the PLIM method with more traditional-type hydraulics methods, with the former taking care of the convection phenomena and the latter making the coupling between the hydraulics equations stronger. A new accurate way of forming the matrices used in the model has also been implemented and these new features are expected to improve the program's robustness, make the treatment of flow channel boundary conditions easier and reduce the importance of numerical parameters given in the input, thus facilitating the practical use of the solver.

International research co-operation

Because the resources of a single country are limited, research in reactor physics and dynamics is very much dependent on international co-operation, where the results of the work are presented e.g. in the form of benchmark studies and other comparisons of measurements and computational results [11], at conferences and in international publications. Finnish participation in for instance the work of NEA, IAEA, AER, NKS, and other Nordic co-operation is also included in the project.

During the first two years of the SAFIR programme, two international conferences and one seminar have been arranged within the EMERALD project. The 11th meeting on "Reactor physics calculations in the Nordic countries" was held in April 2003 and the 14th "Symposium of AER on VVER Reactor Physics and Reactor Safety" in September 2004. The former gathered 46 participants representing 6 organizations in 6 countries, the latter 73 participants from 11 countries (Figure 9). Somewhat smaller-sized was the seminar on "NKS-R 3D Transient Methodology for the Safety Analysis of BWRs" just before the Nordic reactor physics meeting. Naturally, the results of the project were quite extensively presented at these three occasions on its own home ground and the proceedings have been published by VTT [3], NKS in Denmark [1], and KFKI in Hungary [9], respectively.

Applications

As the code system that is the objective of the project will cover the whole range of calculations, from handling of basic nuclear data, i.e. cross section libraries, over fuel and core analyses in normal operating conditions to transient and accident studies, it is to be used for research as well as the needs of the safety authorities and power utilities in order to ensure safe and economic use of the nuclear power plants. Correct evaluation of neutron cross section data is the basis of everything else, static core calculations are needed for in-core fuel management, but also to provide starting-points for transient analyses, and the dynamics calculations aim at identifying and avoiding situations that could jeopardize the safety of the nuclear installations.

Figure 9. More than 70 people from 11 countries participated in the 14th Symposium on VVER reactor physics and safety at VTT.

Conclusions

During the first two years of the EMERALD project, VTT Processes´ computer code system for both steady-state and transient reactor analysis has been further developed and validated through international benchmarks and other comparisons. Cross section libraries and methods for criticality safety calculations have been investigated. Monte Carlo methods have been utilized for different purposes and necessary tools have been acquired to make the calculations and the handling of input and output more efficient. Code development has been focused on nodal and deterministic transport methods. In reactor dynamics, efforts have been made to develop program versions that can cope with problems that the previous codes are not equipped for, especially coolant flow reversal in flow channels.

As its predecessors, 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. The reactor physics work within EMERALD has resulted in a Master´s (criticality safety) and a Licentiate´s (cross sections) thesis. A Doctor´s thesis on the development and validation of MultiTrans is also being prepared.

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2.2 A systematic study of cross section library based discrepancies in LWR criticality calculations

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Abstract

Discrepancies in fundamental nuclear data pose a potential source of uncertainty in neutron transport calculation. These discrepancies are often overlooked in reactor physics calculations. There is evidence that differences in the order of 1% in the multiplication factor are encountered when criticality calculations are carried out using different neutron cross section libraries. Monte Carlo transport calculation codes are especially prone to such discrepancies, since the base evaluated data can be used without modifications.

This paper presents some results of a study, in which cross section library-based discrepancies in light water reactor criticality calculations were investigated in a systematic manner. Point-wise cross section libraries were generated from the ENDF/B-VI.8, JEFF-3.0, JENDL-3.3, JEF-2.2 and JENDL-3.2 evaluated nuclear data files using the NJOY-99 nuclear data processing system. The comparison calculations were carried out using the MCNP4C Monte Carlo transport calculation code. A systematic method based on the neutron balance of the system was developed in order to study the origin of the reactivity discrepancies. The energy dependence of the cross section data was taken into account by dividing the neutron flux spectrum into four energy groups. The comparison calculations cover the most typical LWR operating conditions. The basic geometry is an infinite pin-cell lattice. The main free parameter in the system is the fuel-to-moderator ratio. Several variations of the basic geometry were studied, including lattices with burnable absorber and control pins, a finite lattice with leakage and lattices with low- and high-burnup fuel pins.

Introduction

All calculations in reactor physics require some information on the fundamental interactions between neutrons and target nuclei. These interactions are described by the simple conservation laws of classical kinematics together with complex nuclear physics based on quantum mechanics. The microscopic cross sections describing the interaction probabilities are complex functions of the energy of the incident neutron. A large number of experimental measurements together with theoretical nuclear models are needed to resolve the data. This process, known as cross section evaluation, is an enormous task requiring large capital and human resources.

Currently there are three major evaluation projects in the world: the American ENDF/B, the Japanese JENDL and the JEFF project co-ordinated by the OECD/NEA Data Bank[1]. The evaluated data is publicly distributed in the shared ENDF file format [1]. The major data files are not entirely independent. Knowledge and both measured and evaluated data are shared between the projects. An example of the shared efforts is the Working Party on International Evaluation Co-operation (WPEC) [2], which was established to develop the quality, completeness and consistency of the data.

Despite the international efforts, there are still differences between the evaluated data files. The inconsistencies result from the fact that there are always uncertainties in experimental measurements and free parameters in theoretical nuclear models. The discrepancies in the data are inevitably reflected to all reactor physics calculations. This problem concerns especially Monte Carlo transport calculation codes, which are able to use the data directly, i.e. in a point-wise form without major modifications. Deterministic codes, on the other hand, use very problem-specific group-wise data, which is typically calibrated to give good results in the application environment.

Various comparison calculations have shown that discrepancies in the base nuclear data may result in differences up to 1% in the multiplication factor. Such differences can be over 10 times larger than the statistical error of the Monte Carlo calculation method and almost comparable to the safety margins in criticality safety studies. Cross section libraries based on the different releases of ENDF/B-VI tend to under-predict the multiplication factor in typical LWR systems [3, 4], while libraries based on JEF-2.2 and JENDL-3.2 usually give slightly higher values. It has also been shown that the discrepancies between the libraries are significantly larger in under-moderated systems [5]. Although most of the comparison calculations have been carried out using the MCNP code, similar differences have been encountered using other Monte Carlo codes as well [6].

The impacts of nuclear data discrepancies on the results of basic reactor physics calculations have been systematically studied at the Technical Research Centre of Finland (VTT). This paper presents some of the results of the recent studies, which have been published in full as a VTT Processes project report [7].

MCNP criticality calculations on simplified LWR core models were carried out using different cross section libraries. The comparison of results reveals large variation in the multiplication factor. The differences are systematic and strongly dependent on the level of neutron moderation. An analytic method based on the neutron balance of the system

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[1] The JEFF data file is mainly built on two previous European projects, the Joint Evaluated File (JEF) and the European Fusion File (EFF).

was developed in order to study the origin of the reactivity discrepancies. The neutron spectrum is divided into separate energy groups, so that the discrepancies in the different energy regions of the data can be revealed.

Tools and methods

The comparison calculations were carried out using the MCNP4C particle transport code developed at Los Alamos National Laboratory [8]. The cross section data for MCNP were generated using the NJOY-99 nuclear data processing system [9]. Five evaluated nuclear data files were used in the calculations. Most of the publicly available cross section libraries for MCNP are based on data from ENDF/B-VI. For this reason, the latest version of this data file, ENDF/B-VI.8 [10] released in 2001, was taken as the reference case. Results based on two recently released data files, JEFF-3.0 [11] (April 2002) and JENDL-3.3 [12] (May 2002), were compared to the reference results. Two older but still widely used evaluations, JEF-2.2 [13] (1993) and JENDL-3.2 [14] (1994), were also included.

The geometry of the system consists of uranium oxide fuel pins in light water moderator. The pins are arranged in an infinite square lattice. The fuel is enriched to 4.0 at-% U-235. The cladding is made of pure natural zirconium. All minority isotopes in the fuel, cladding and moderator are omitted. The main free parameter in the system is the fuel-to-moderator ratio (FMR), which was varied by changing the moderator void fraction. All materials are at room temperature (300 K). The pin-cell geometry is illustrated in Figure 10.

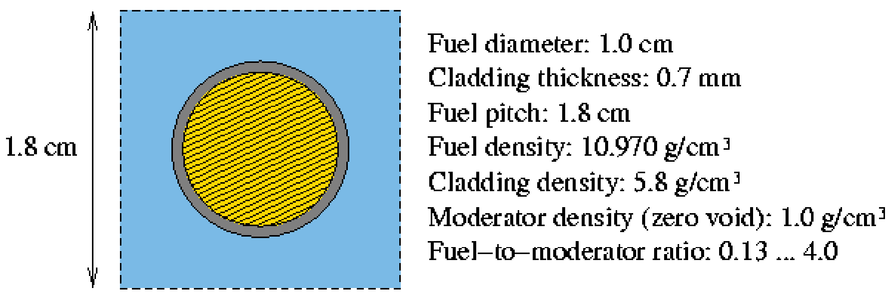

Figure 10. Description of the pin-cell model.

The results presented here include only calculations carried out using the regular pin-cell lattice described above. The calculations in the study also included various modified lattices. Discrepancies in finite lattices with leakage, lattices with burnable and control absorber pins and lattices with burned fuel were investigated. The detailed results can be found in Reference [7].

The fuel-to-moderator ratio was varied from 0.13 to 4.0, which covers most of the typical LWR operating conditions. When the moderator density is reduced, fewer neutrons are scattered into the thermal energies. Depending on the shape of the flux spectrum, differences in the various energy regions of the cross section data are pronounced.

A direct comparison of multiplication factors given by the MCNP criticality calculation shows the order of magnitude of the difference between the libraries. In order to trace back the origins of the observed discrepancies, an analytical method based on the detailed neutron balance of the system was developed. The infinite multiplication factor can be written as the ratio of simple source and sink terms:

$$k_\infty = \frac{F}{S_F + S_C + S_M}, \tag{1}$$

where F is sum of the fission source and the (n,2n) scattering source and S_F, S_C and S_M are the absorption terms in the fuel, cladding and moderator, respectively. Each source and sink term consists of the corresponding reaction rates, which may be divided into separate energy groups. The source and sink terms in the fuel region, for example, can be written as:

$$F = \sum_{g=1}^{4}\left[V_F \Phi_F^g \sum_m \left(\overline{\nu}_m^g \Sigma_{f,m}^g + \Sigma_{2n,m}^g\right)\right]$$

and

$$S_F = \sum_{g=1}^{4}\left[V_F \Phi_F^g \sum_m \left(\Sigma_{f,m}^g + \Sigma_{\gamma,m}^g + \Sigma_{\alpha,m}^g\right)\right]$$

The inner summations run over all materials in the fuel[2] and the outer summation over four energy groups. The key idea of this representation is that all the reaction rates can be calculated very easily using the standard cell flux tallies of MCNP.

The sensitivity of k_∞ against small changes in individual reaction rates can be estimated by linearising Equation (1) with respect to the corresponding parameter. If reaction rate x is deviated from its reference value by Δx, the estimated change in k_∞ can be written as:

[2] Some of the terms, such as the fission and the (n,2n) cross sections of fuel oxygen, are zero or practically negligible.

$$\Delta k_\infty = \Delta x \frac{\partial k_\infty}{\partial x}\bigg|_{\text{ref}} \tag{2}$$

The partial derivative is calculated in the reference spectrum, i.e. using the results given by the reference calculation.

Once the reaction rates are calculated for each library, Equation (2) gives a simple means to estimate how particular differences in the cross section data might affect the total differences in the multiplication factor. Since the absolute value of k_∞ varies over a wide range as the fuel-to-moderator ratio is changed, it is more convenient to make the comparisons for reactivities:

$$\rho = \frac{k_\infty - 1}{k_\infty}.$$

This does not significantly complicate the related equations. The formulation of the partial derivatives is a matter of simple calculus.

Results

The results of the criticality calculations are plotted in Figure 11. It is clear that there are large discrepancies between the libraries and that the differences grow as the level of neutron moderation is reduced. In typical light water reactors, the local FMR varies from about 0.2 to 1.7, depending on the reactor type and the local thermohydraulic conditions. It can be seen that both older libraries, JEF-2.2 and JENDL-3.2, give approximately 500–1000 pcm higher values in this region, when compared to the reference results. The more recently published JEFF-3.0 and JENDL-3.3 are more consistent. Very similar results have been obtained in various comparison studies using realistic geometry models of actual criticality experiments [3, 4, 5, 6].

It can be seen in Figure 11 that the reactivity discrepancies are systematic and strongly dependent on the neutron spectrum. As the level of neutron moderation is reduced, reactions in the thermal energy region become less significant. The thermal fission of U-235 is replaced by U-235 fission in the resonance region (energy group 3) and by the fast fission of U-238. Parasitic neutron absorption increases in the third and the second energy group. The cross section curves become more complex at higher energies. It is clear that there are more differences in the data as well.

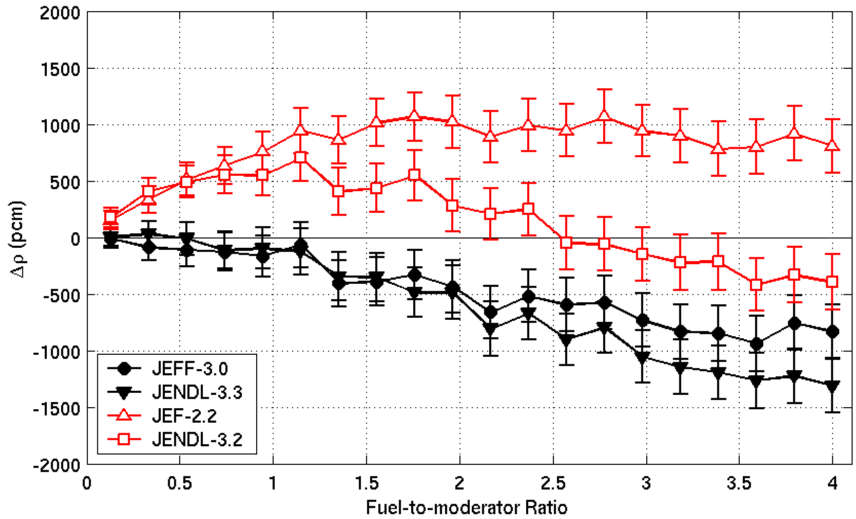

Figure 11. Library-based differences in reactivity as function of fuel-to-moderator ratio. Comparison to ENDF/B-VI.8.

The contributions of individual reaction rates can be evaluated using Equation (2). For the older libraries (JEF-2.2 and JENDL-3.2), there are several terms with significant contributions in the overall result. The differences for JEFF-3.0 and JENDL-3.3, on the other hand, originate predominantly from two reactions: the fast fission and the resonance capture of U-238. Some of the differences result from small discrepancies in dominating terms, while others result from large differences in less significant reaction rates. The overall consistency between two libraries mainly depends on how well the individual factors cancel each other out.

It is not surprising that the reactivity discrepancies mainly originate from the uranium isotopes. There are very large differences in the absorption rate of fuel oxygen, but the reactivity contribution of this reaction is rather minute. This is also the case for many isotopes in the burnt fuel calculations [7]. There are several actinide and fission product isotopes in the fuel, but the main reactivity contributors are the uranium isotopes together with Pu-239. Neutron leakage, burnable absorbers (Gd) or boron carbide control pins had no significant impact on the results compared to the base case. Some additional reactivity discrepancies were, however, discovered using control pins containing AIC absorber.

Conclusions

The role of cross section data as a source of uncertainty in LWR criticality calculations was investigated in a systematic manner. The NJOY-99 nuclear data processing system was used to generate point-wise cross section libraries from the ENDF/B-VI.8, JEFF-3.0, JENDL-3.3, JEF-2-2 and JENDL-3.2 evaluated nuclear data files. Comparison calculations using simplified LWR pin-cell lattices were carried out using the MCNP4C Monte Carlo transport calculation code. An analytical method based on the detailed neutron balance of the system was developed in order to study the sources of the observed reactivity discrepancies.

The main free parameter in the calculations was the fuel-to-moderator ratio, which was varied by changing the moderator density. It turned out that this parameter has a very significant impact on the results. The differences between the libraries are large and systematic. In typical LWR operating conditions, the discrepancies between the older libraries (JEF-2.2 and JENDL-3.2) and the reference ENDF/B-VI.8 results are in the order of 500–1000 pcm. The more recently released JEFF-3.0 and JENDL-3.3 give more consistent results. The differences grow significantly as the level of neutron moderation is reduced.

The differences in reactivity result mainly from the fission and the radiative capture cross sections of U-235 and U-238, especially in the resonance region. These isotopes are the main sources of uncertainty also in calculations involving burnt fuel or modified lattices with absorber materials. A common feature to all cases is that the overall reactivity discrepancies mainly depend on how well the individual contributing factors cancel each other out.

Sensitivity studies performed on the main parameters [7] showed that the results are not significantly affected by small changes in fuel and moderator temperature, fuel enrichment, moderator boron concentration or the level of heterogeneity of the geometry model. Nevertheless, the results of this study should not be directly generalised to more complex cases, in which there are various additional factors contributing to the neutron balance of the system. Instead, the results should be reviewed as a simplified example of the uncertainties that may lie hidden within the cross section data. The discrepancies in the data should be treated as any other potential source of error in reactor physics calculations.

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2.3 The coupled code TRAB-3D-SMABRE for 3D transient and accident analyses

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Abstract

The three-dimensional TRAB-3D core dynamics code is being internally coupled to the thermal hydraulics system code SMABRE. The codes have previously been coupled with a parallel coupling scheme. VTT's reactor dynamics codes have performed well in all the situations that they have originally been designed for. The most important limitation of the present code models is their inability to handle coolant flow reversal in the core channel, a phenomenon that can be encountered in e.g. BWR ATWS cases or VVER power excursions. The new coupling of the two codes is realized on the level of each node of each channel in the core, with each fuel bundle described with its own channel. Necessary interfaces have been created, an improved version of SMABRE's thermal hydraulics solution method developed, and a steady state procedure developed. A satisfactorily working steady state solution has been achieved. The next step in the development will be testing of the transient calculation. Besides solving the flow reversal limitation of the present dynamics models, a successful coupling will allow expanding into more realistic modelling of an open core.

Introduction

TRAB-3D [1] is a reactor dynamics code with three-dimensional neutronics coupled to core and circuit thermal hydraulics. The code can be used for transient and accident analyses of boiling (BWR) and pressurized water (PWR) reactors. The system code SMABRE [2] models the thermohydraulics of light water reactors. Both codes have been entirely developed at VTT.

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. For PWR applications the TRAB-3D and SMABRE codes have been coupled earlier by using the parallel coupling principle: the full core TRAB-3D hydraulics coupled to neutronics and heat transfer, and the coarse SMABRE core hydraulics with fewer channels are solved in parallel. The rest of the circulation system is solved with SMABRE. The solution method of the SMABRE circuit model is non-iterative and only a loose coupling with no iterations exists between the two codes.

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.

VTT's dynamics codes have performed well in all the situations that they have originally been designed for. The most important limitation of the present code models is their inability to handle coolant flow reversal in the core channel, a phenomenon that can be encountered in e.g. BWR ATWS cases or VVER power excursions. To remove this limitation, the TRAB-3D neutronics and the SMABRE thermal hydraulics code are being coupled together using an internal coupling scheme. In the new concept TRAB-3D will perform only the neutronics and fuel pellet heat conduction and SMABRE will take care of the hydraulics calculation of the whole cooling circuit including the reactor core, and heat transfer on the cladding surface.

The TRAB-3D code

TRAB-3D is an independent BWR dynamics code with rectangular core geometry and with 3D neutronics coupled to parallel 1D channel hydraulics. In the core the parallel 1D components are applied for all individual fuel elements, and the rest of the BWR thermohydraulics is described by using 1D components. The code uses a fast, sophisticated, two-level iteration nodal method. The implicit time discretization allows flexible time step choices. The BWR circuit model originates from the TRAB-1D models [3]. The core can be modelled using the 1/1, 1/2 and 1/4 core symmetries.

The core model can be used separately for PWR or BWR calculations. These features enable coupling of the code with an external thermohydraulic model.

The one-dimensional TRAB-1D code has been extensively used for the plant analyses of the Finnish TVO reactors of BWR type. The three-dimensional TRAB-3D has been validated against OECD LWR core transient benchmarks, and real plant transients for the Olkiluoto 1 plant, including pump trip, pressurization transient, instability incident and load rejection test including partial asymmetric scram. Validation of the code is summarized in [4]. TRAB-3D is now in production use for plant transient and accident analyses.

The SMABRE code

The thermohydraulic model of SMABRE describes the physics of the two-phase mixture of liquid and gas. The code is based on a node-junction hydraulic circuit model. The five equation formulation with the drift flux phase separation is modelling the two-phase behaviour. Conservation equations are solved for the phase mass, mixture momentum and phase energy. Additional equations are for the noncondensables in gas and boron in liquid.

The transient is controlled by boundary conditions defining the mass, momentum or energy sources. SMABRE simulates the main sections of the primary and secondary circuit. The pressurizer valves, primary letdown, steam line relief valves and turbine valves are the boundary conditions as outflows. The charging flow, emergency core cooling, accumulator, main feedwater and auxiliary feedwater are the boundary conditions as inflows. The boundary conditions, originally aimed for the LOCA simulation, are in principle enough for many transients, but in other transients more sophisticated models may be needed for the recirculation pump speed, feedwater injection and turbine valve opening control.

The validation of SMABRE includes mainly calculations related to tests in integral facilities, which were often arranged as international standard problems by the OECD/CSNI. Validation cases of SMABRE are listed in e.g. [5].

As compared to large system codes, like the RELAP5, CATHARE, and the ATHLET, SMABRE has a limited simulation capability, but by concentrating on the most important modeling aspects around the LWR safety the code can be considered as a rather versatile analysis tool. By using SMABRE in the combined products in the form of simulators and integral codes, the analysis of many such transients and accidents is possible, which are outside the applicability range of the traditional system codes.

Neutronic-thermohydraulic coupling

Three different modes for coupling dynamic core models and system codes are utilised worldwide, called external, internal and parallel coupling. With external coupling the whole core calculation is carried out by the dynamics code, and the system code is used for the rest of the circuit. This approach has not been applied at VTT.

The first realized coupling concept at VTT is parallel coupling, the principles of which are illustrated in Figure 12. In this mode the two coupled codes are running independently with minimum amount of data transfer between the modules. Thermal-hydraulics of the core is calculated with both codes in parallel, but rest of the circulation system is solved with SMABRE. The connection is carried out by data exchange once in a time-step for core inlet flow and outlet pressure into the TRAB direction and core power distribution into the SMABRE direction. The first application realised with this principle was for the 1-D neutronics in 1988 and for the 3-D neutronics in 1991–1992. VTT has more than a decade of experience in carrying out safety analyses with coupled three-dimensional codes. The TRAB-3D / SMABRE parallel coupling was validated against the OECD benchmark calculation of main steam line break transient in the TMI-1 PWR plant [6].

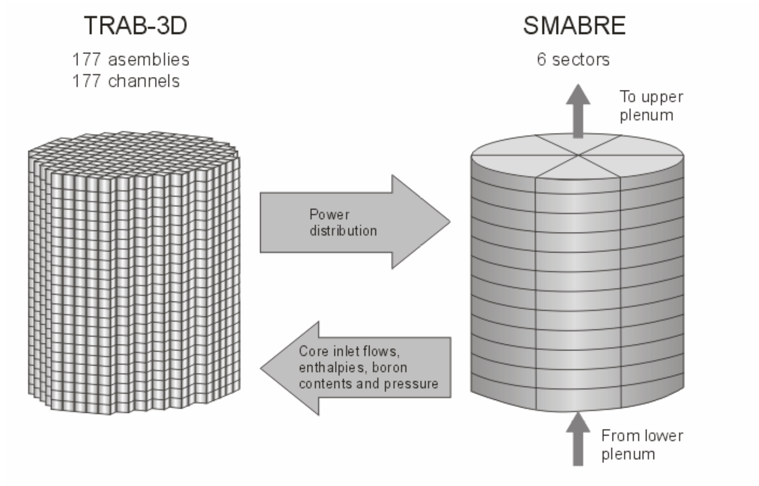

Figure 12. The parallel core coupling principle.

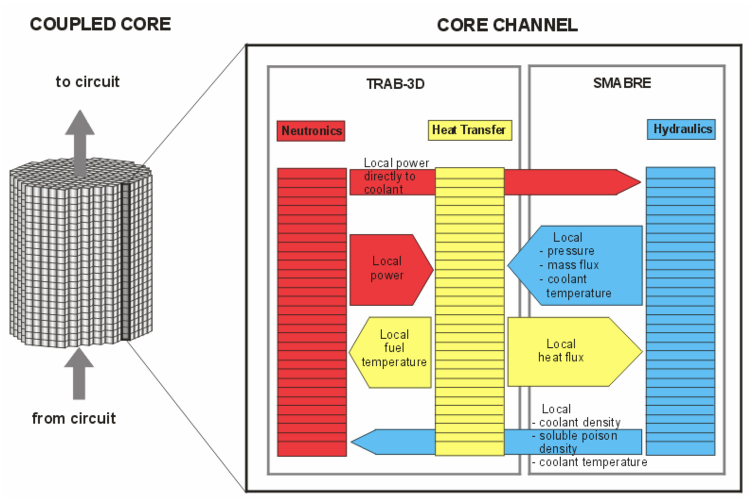

Figure 13. The internal core coupling principle.

In Figure 13 the internal coupling principle has been illustrated. The interface between modules is realized inside the neutronic solution itself, and update is done inside the TRAB iteration procedure. The coupling is rather complex, and required a lot of modifications into the internal solution procedures of both codes. Finally it was found as the best solution that TRAB delivers for SMABRE the cladding temperature on each time step and direct heat absorption into the coolant. Based on this information SMABRE calculates the heat flux on the cladding surface and based on the entire loop model the reactivity parameters, coolant density and boron concentration for the neutronics. They are applied in the new TRAB iteration until the iteration converges. TRAB itself calculates the Doppler effect of fuel.

The SMABRE thermohydraulics is used for all core channels. Testing and adaptation of the matrix solver of SMABRE for modelling with a large number of core channels produced three alternative approaches, which result in shorter calculation time than the original solution. Each core channel is solved with the band matrix solution and the subchannels solutions are coupled iteratively together at the ends of the core channels. The solution has been tested with 500 parallel channels and 25 axial nodes, corresponding to a reference BWR core. For the PWR plants the method is used as well by simulating the core by one-dimensional parallel channels.

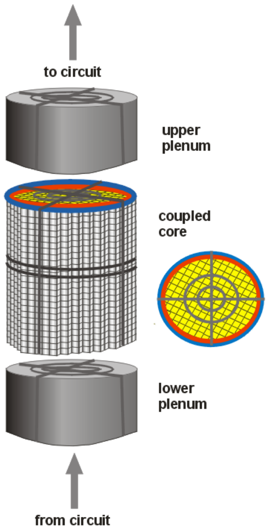

Figure 14. The coupling of the vessel subdivision with the core model.

The principles of connecting the circuit model with the core model are illustrated in Figure 14. The reactor vessel may be divided into the sectors for the asymmetric flow calculation. In the core inlet several rings and sectors may exist, and each section is connected with individual fuel and flow channels. This allows taking into account some transverse features in the thermal hydraulics calculation. New development is needed later for the fully open core calculation with cross-flow flow connections, e.g. to couple core neutronics to a thermal hydraulics porosity type model.

Presently the codes have been coupled and the necessary modifications implemented. The hydraulics solution of TRAB-3D has been separated from the rest of the calculation. TRAB-3D heat transfer calculation has been modified so that the calculation of heat transfer from cladding to coolant has been transferred to SMABRE. The subroutine interface connecting TRAB-3D and SMABRE has been created, with connection data exchanged in every channel and in every node of the core. A procedure for creating the core geometry for SMABRE from the existing TRAB-3D input has been completed.

Following originally developed TRAB-3D principles, the core calculation contains the outer and inner iteration level. During the outer iteration the converged solution is searched for the core thermohydraulics and neutronics. In the initial solution both of these were iterative. In the new solution the iterative neutronic solution has been maintained, but the thermohydraulics results from the repeated SMABRE integration over the timestep. Filtering coefficients are used for the core inlet conditions, but the channel itself is solved by the band matrix inversion. Inside the outer iteration SMABRE time integration is repeated until the neutron flux distribution has been converged. During each outer iteration the new neutron flux corresponding to the core void, fluid temperature and fuel temperature distribution is searched. The heat conduction equation is integrated with new heat transfer and power generation during the iteration step, and as a consequence the core heat flux distribution will be changed.

The rather sophisticated TRAB features for describing PWR and BWR bundle geometry and fuel compositions are maintained. Each bundle is related to an individual flow channel, flow channels may have axial subdivisions, and each fuel element may have an individual fuel rod composition. The SMABRE nodalization enables multidimensional subdivision in the lower and upper plenums through circumferential sectors and radial rings.

Testing of the coupled code

Recent work has been mostly concentrated on creating a stable solution for the initial steady state. The physical processes in the core are inherently coupled. Therefore it has been necessary to realize the coupling inside the TRAB solution procedures, including SMABRE's thermohydraulic solution inside the iterations as well. TRAB solutions were designed for coupled iterations, and were not originally well adjusted to externally calculated thermohydraulics. This is why in parallel with getting a stable numerical solution, large efforts were needed simply for coupling the different solution philosophies of the two codes together. This is especially true in searching of the steady state, which in TRAB is done iteratively, while SMABRE has proceeded to a desired state by calculating forward in time until it is achieved.

In the finalized version TRAB can be run in three possible modes: TRAB alone, parallelly coupled and internally coupled. This possibility has been important for comparison of steady state conditions. At present the stationary solution is functioning satisfactorily, and SMABRE calculates the core pressure distribution as the original TRAB. The next step is calculation of transient cases, first by introducing isolated disturbances, comparing TRAB-3D and TRAB-SMABRE, and proceeding to complex transients comparing against plant data.

Sophisticated plant specific models exist for the recirculation pumps, steam separator, feedwater dynamics, turbine controller valve dynamics, and pressure pulse propagation in the steam line. The comparison of the similarity of these features between the original models of the two codes is in progress. Some additional data transfer for transient calculations with the coupled code is also needed (e.g. trip data from TRAB to SMABRE).

After these adjustments the behaviour in the flow reversal will be examined. The applications leading to these conditions will be searched after finalisation of the first testing phase. At least RIA, ATWS, and abrupt MSIV closure are considered for BWR plants and RIA and ATWS for PWR plants.

Conclusions

The three-dimensional TRAB-3D core dynamics code has been successfully coupled to the thermal hydraulics system code SMABRE, and a satisfactorily working steady state solution has been achieved. The coupling has been more laborious than could be foreseen, due to the inherently coupled nature of the physical processes and the different solution philosophies of the two codes. The next step into testing of the transient calculation is expected to be less complicated after the steady state procedures have been found to function well enough.

Besides solving the flow reversal limitation of the present dynamics models, a successful coupling will allow more realistic modelling of an open core. It will open up new options to couple the core model to other thermal hydraulic system codes, and enable further work to couple core neutronics to a thermal hydraulics porosity type model.

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3. High-burnup upgrades in fuel behaviour modelling (KORU)

3.1 KORU summary report

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Abstract

The modelling in fuel behaviour codes that are in use at VTT are being upgraded to meet the requirements from evolving fuel design and operational data – most notably higher burnup goals – and from revised guidelines for applying the licensing procedures. Work is being carried out to partially renew the descriptions of fission product swelling and release, and detailed mechanical response. As effective probabilistic methods will be favoured even in fuel accident behaviour codes, bases for such studies are being founded. Emphatic education and training of a new generation of experts is showing good progress.

Introduction

Two separate lines of fuel behaviour codes are operative and vividly used for applications at VTT. Within both, however, descriptions of fuel behaviour phenomena do not meet requirements in all respects when markedly higher burnup ranges need to be covered.

In steady-state codes, details of fission gas release models are being reviewed and elaborated to better reproduce the experience from high-burnup operation and testing. Detailed knowledge of high-burnup fuel behaviour in Loss-of-Coolant and Reactivity Initiated Accident conditions is not sufficient. The Finnish utilities, STUK, and VTT are jointly well placed in internationally managed efforts such as the OECD-IRSN CABRI Water Loop Project, the experiments in the OECD Halden Reactor Project, and the work in the US National Laboratories ANL and PNNL. Model development and validation will proceed along with accumulating representative data. Earlier VTT development resulted in introducing a combined model – FRAPTRAN-GENFLO – in which a detailed fuel transient model has been interactively coupled with an advanced flow model for handling complicated thermal hydraulic conditions. Applications and further validation continue.

Related working groups and programmes within international organisations are attended.

Efforts for more efficient fuel utilization will continue both in the existing Finnish plants and with the fifth reactor being now under construction. These undertakings particularly accentuate the importance of having up-to-date independent modelling tools timely available.

Main objectives

Objectives over the whole programme period include:

- Elaboration and validation of FRAPTRAN-GENFLO – a code with advanced thermal hydraulics combined with detailed fuel description – for versatile applications

- Introduction of a mechanistically based fission gas release model for the ENIGMA code. More generally, extension of the modelling of the codes to be valid for 55 to 65 MWd/kgU rod burnups. Probabilistic methods will be increasingly favoured

- Acquiring data from international experiments on Loss-of-Coolant and Reactivity Initiated Accidents and on emerging new materials to upgrade the performance models

- Establishing improved models for mechanical behaviour and failure modes

- Supporting education and training of a new generation of experts in the field.

Main results

The combined FRAPTRAN-GENFLO code has been adapted to describe the Halden Project LOCA test rig. The first two preliminary tests have been successfully analysed. The planning has included studies to estimate the effects of several test parameters (Figure 15). Comparison with observed behaviour is convincing (Figure 16.)

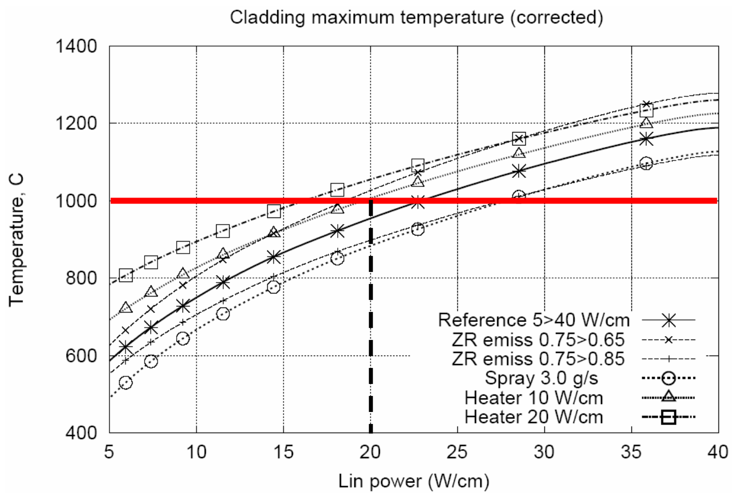

Figure 15. Effects of parameters in a Halden IFA-650 LOCA test according to FRATRAN-GENFLO.

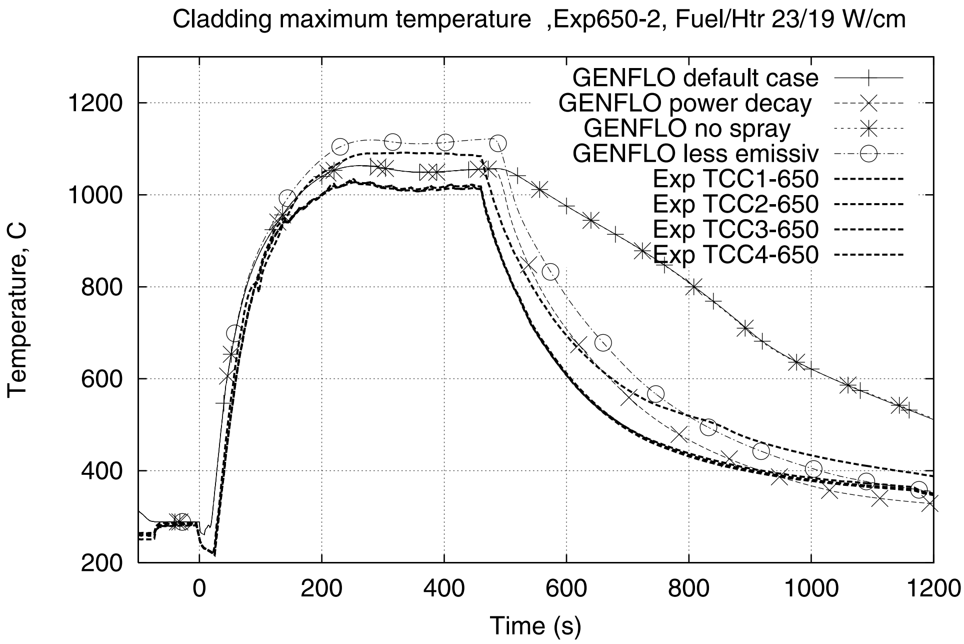

Figure 16. Comparing LOCA test temperature measurements with calculated GENFLO code results.

Further support has been provided for the planning of the first test to be run with an actual high-burnup rod.

The fission gas release model in the ENIGMA code has been tested and re-evaluated. Use has been made of the data base and test cases of the IAEA FUMEX II Co-ordinated Research Programme and comparison with other codes (Figure 17). The high-burnup low-temperature part of the description needs improvement. A collaboration work involving an outside consultant has been established to acquire an optional mechanistic fission gas release model. Implementation of the code is under way.

Reviewing probabilistic methods has continued to introduce core-wide mapping of fuel performance in transient and accident conditions. Such a procedure will be instrumental in verifying license requirements of estimating the number of failing rods in an accident.

An extensive effort was carried through to attach a finite-element based model in fuel performance codes. This involves options to proceed to two and three dimensional descriptions and to allow handling pellet and cladding creep, pellet-to-clad frictional slip, and high cladding deformations such as ballooning (Figure 18). Most of the basic module up to a two-dimensional description has been completed with encouraging examples of the advantages of the new approach. The introduction of the new solution was made into the USNRC FRAPCON-3 and FRAPTRAN codes. The addition is, however, modular and readily applicable in other codes.

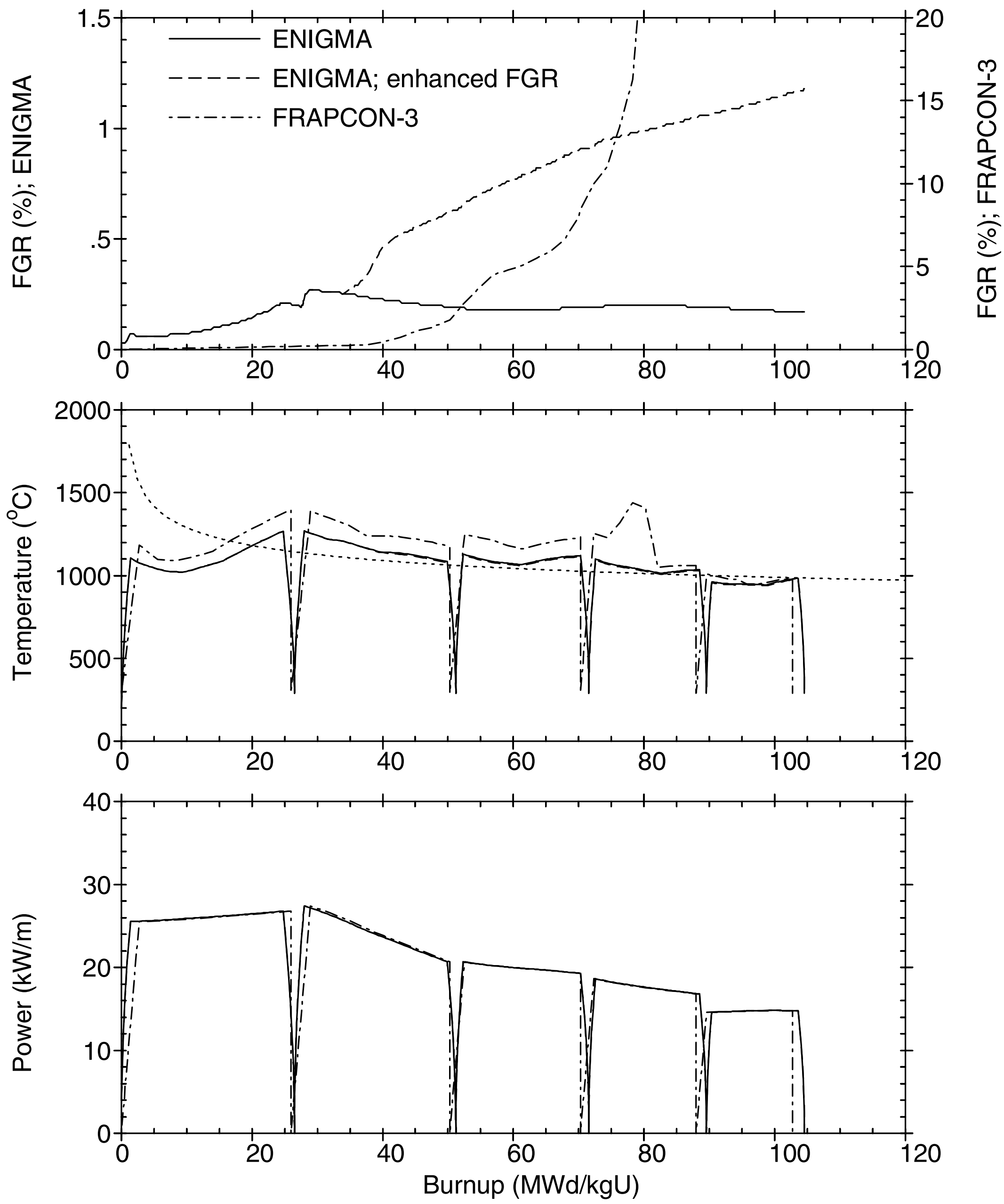

Figure 17. Comparison of ENIGMA and FRAPCON-3 fission gas release results in a standard test case.

A VTT scientist has worked at the PNNL laboratory in the US for one year (see FRAPCON-3 and FRAPTRAN code development above and in a special article), with a draft for a licentiate thesis as a further result. VTT undergraduate trainees have attended several short courses of fuel performance related contents.

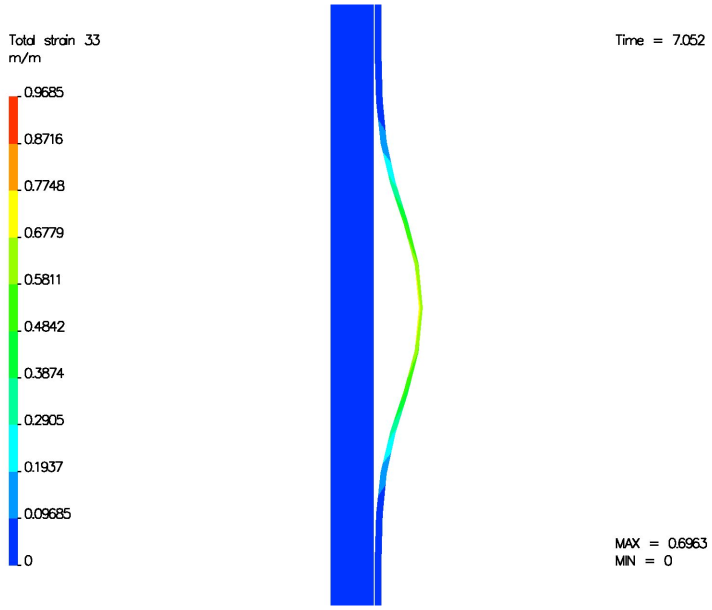

Figure 18. Large cladding deformations as described with the newly-developed FEM based formulation.

In 2004, Finnish participation in the OECD-NEA Studsvik Cladding Integrity Project was prepared. Funding will be from the Finnish utilities, with VTT serving as a coordinator of participation and applications.

Conclusions

Fuel performance modelling remains a challenging field in reactor technology. Evolving effectiveness and regulatory requirements in one hand, and novel materials and operational practices on the other, prompt continuing research efforts in the area.

The progress suggests that the goal of preserving an independent up-to-date domestic capability for all the fuel performance studies that may be required in the near future is approached. Coordinating the public research among VTT, STUK and the utilities, and well placed international collaboration facilitate effective distribution of the limited financial and human resources.

3.2 Improvements on FRAPCON3/FRAPTRAN mechanical modelling

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Abstract

Work is described that has been done to modernise the mechanical modelling of FRAPCON and FRAPTRAN codes. FRAPCON and FRAPTRAN are nuclear fuel performance codes for steady-state and transient analyses, respectively. A new stress and strain analysis approach with a finite element model has been implemented in the codes. The finite element model is capable of handling material nonlinearities such as elastic, thermal, plastic and creep deformations, and geometric nonlinearities like large localised cladding deformations, e.g. ballooning of the cladding under loss-of-coolant accident conditions.

Introduction

FRAPCON and FRAPTRAN are fuel performance codes for steady-state and accident analyses. FRAPCON is a steady-state code, which simulates the thermal mechanical behaviour of a single fuel rod in normal reactor operation. FRAPTRAN is a transient code for light water reactor (LWR) fuel rods and it can be used to simulate the fuel rod performance in design basis accidents, e.g. in loss-of-coolant accident or LOCA and in reactivity initiated accident or RIA. The codes have been developed by the US Nuclear Regulatory Commission (USNRC), and currently the development and maintenance of the codes are done at the Pacific Northwest National Laboratory (PNNL) under USNRC funding. The codes are used for licensing and safety analyses of LWR fuel rods. The codes are property of the USNRC. The USNRC has organised users' groups through which the codes have been made available and they are used in many safety and research organisations around the world. The codes are applied to safety-related fuel performance analyses also at the Technical Research Centre of Finland (VTT). Over several years VTT has participated in the development work of these codes. One of the results of this previous development work a VTT is the coupling of an advanced thermal hydraulic model GENFLO with FRAPTRAN. [1, 2, 3]

Both FRAPCON and FRAPTRAN have long development histories, which originate from work started in the 1970's. The codes employed a rather simple 1D thin-shell mechanical model for the stress-strain analysis for the fuel rod cladding. However, there have been significant developments in both solid mechanics and computer hardware since that time. The wide research into numerical techniques such as finite element methods and increased computer power have made it possible to use large scale numerical models, which take into account both material and geometric nonlinearities. In fact, the use of the nonlinear finite element methods is currently the standard method for the stress and strain analysis of structures.

The goal of this work was to implement a state-of-the-art mechanical model in FRAPCON and FRAPTRAN, which would be able to handle deformation mechanisms at normal operation and under postulated accident scenarios. Much of the code is based on the finite element solver that has been implemented earlier at VTT. [4] The implemented mechanical model uses finite element formulation to solve the stresses and strains in the cladding of a fuel rod. Currently the codes do not have stress-strain analysis in the fuel pellet stack. However, the new finite element model is designed to be flexible and stress-strain analysis also in the fuel pellet stack may be implemented later with little additional effort. The finite element implementation is also designed to be flexible to merge simple 1½-dimensional modelling with full 2D and 3D modelling for better numerical efficiency. For example, in a general LOCA case the cladding of a fuel rod may experience a localised ballooning in a short axial part whereas the deformations in the rest of the rod remain fairly axisymmetric. The goal of this work is to use simple, computationally economical, 1½-dimensional modelling for the most of the fuel rod and use a 2D or a 3D mesh only where truly needed.

Finite element model

Finite element method is a numerical technique that can be used to solve partial differential equations or PDEs. Typically there is no analytical solution for the problem because of the complexity of the physical problems excluding few very simple special cases. For example, in case of solid mechanics, a complex relation between deformations and stresses and complexities of geometry of analysed structures make the use of numerical methods necessary. Probably the most powerful and at least the most used numerical method is the finite element method. The finite element method approximates the solution field of the PDE with a set of discrete values at nodal points. The field between the nodal points is approximated by simple polynomials or shape functions, which approach can be quite easily suited to describe even complex geometries. Nevertheless, in the case of large deformations, suitable formulation of the appropriate PDE and formulation of the relations between the stresses and deformations are challenging tasks.

An updated Lagrangian formulation is employed in the implemented FE model, i.e. FE equations are formulated in reference to the deformed configuration. The model uses logarithmic strains and Cauchy stresses (natural strains and true stresses) in its material models. The finite element model is capable of modelling material nonlinearities such as elastic, thermal, plastic and creep deformations and geometric nonlinearities or large strains. Such large localised deformations can occur e.g. under loss-of-coolant accident conditions (ballooning).

The model employs multiplicative decomposition of the elastic and inelastic deformations, i.e. the total deformation gradient tensor **F** can be factored to elastic and inelastic parts

$$\mathbf{F} = \mathbf{F}_e \, \mathbf{F}_{pl}.$$

The implemented models are also capable of handling a frictional contact between fuel pellets and cladding. The finite element implementation can mix 1½D, 2D and 3D elements in a single model of a fuel rod. In this way, 2D or 3D elements can be used to describe the stress and strain state at the localised region that is undergoing ballooning while a computationally more efficient 1½D mesh is sufficient for the rest of the fuel rod. The FE mesh can be also modified during the calculation. An example of a remesh in the hottest axial zone, where the ballooning is expected, in FRAPTRAN calculation is shown in Figure 19. 1½D mesh is replaced by a real 2D mesh, which is capable of describing the ballooning region.

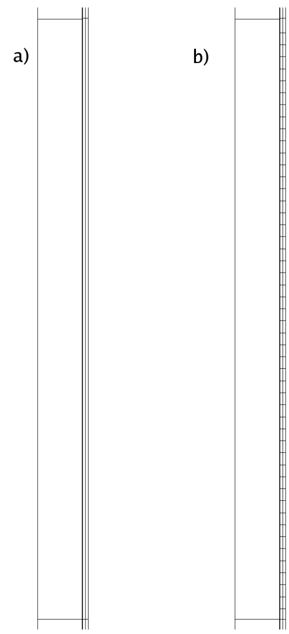

Figure 19. 1½D mesh in (a) is replaced by finer bilinear 2D element mesh (b) in the hottest slice when the ballooning is anticipated.

Linear, bilinear, and trilinear elements are used for 1½D, 2D and 3D cases, respectively. These lower-order elements were chosen because of their robustness in large deformations and distorted element meshes. The shear and volumetric locking problems that are typical of the lower-order 2D and 3D elements were solved by separating the deviatoric and dilational stress and strain tensors and using mean dilation procedure to integrate the dilational parts of the finite element equations.

The chosen nonlinear FE model employs a Newton-Raphson type iterative solution method, with which a nonlinear system is solved by a series of linearisations. The efficiency of this type of solution methods depends on the efficiency of the linear system solver. The implemented linear equation solver uses direct methods (LU and LDL^T decomposition) to solve the symmetric or nonsymmetric linear system. The linear system solver takes into account the sparsity of the system, i.e. zero terms of the linear system matrix are not stored in the computer memory. For example, the solver reorders the linear system in a way that the need for the computer memory is minimized, see Figure 20.

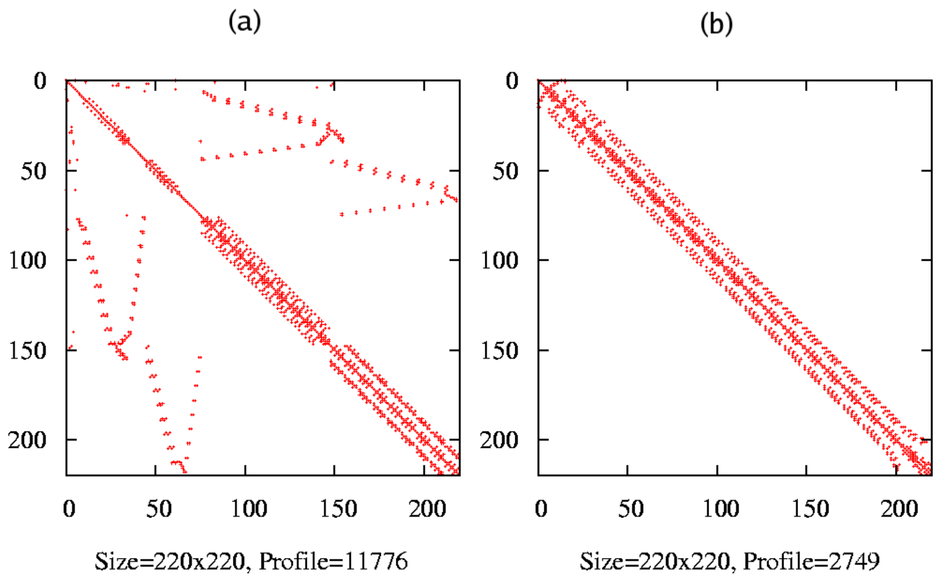

Figure 20. The nonzero structure of sparse matrix before and after reordering. Reordering reduces the needed memory space to store the sparse matrix and its factors by 75% in this case.

The finite element implementation is designed to be flexible for any further improvements. For example, the elements are separate modules, which allows one to create new element modules with minimal changes in the main driver program of the FE model. Attempts have been made to keep the usage of the new models as simple as possible. Using the new stress strain modelling does not require special knowledge on finite element methods or solid mechanics. For example, the meshing is generated automatically and there are only few additional input parameters for the codes that are specific to the new finite element analysis. In addition to numerical aspects of the finite element analysis, also a graphical tool for post-processing the analysis results was created. Although this tool is fairly simple, it allows producing high quality figures from the finite element analysis results. The figures in this report are plotted with the new graphical tool.

Examples of the verification and validation cases

To verify the numerical correctness and robustness of the numerical implementation, a series of verification cases were analysed. FRAPCON/FRAPTRAN implementation does not allow the user to set material properties or define other geometries than the fuel rod geometry itself. However, these simple verification cases are necessary to exclude any obvious flaws in the current implementation.

The modular design of the FRAPCON/FRAPTRAN allows the new models to be tested in a general-purpose finite element solver, NLFEMP, that has been implemented at VTT earlier. [4] The modular design allows for exchange of e.g. elements between NLFEMP and FRAPCON/FRAPTRAN without any changes in the element modules. The element implementation was tested with simple patch tests. Also series of large-deformation elastic-plastic cases were run to verify the robustness of the models in severe test cases. An example of the used large-strain test case, a well documented case by Taylor et al. [5] for elastic-plastic upsetting of an axisymmetric billet is shown in Figure 21.

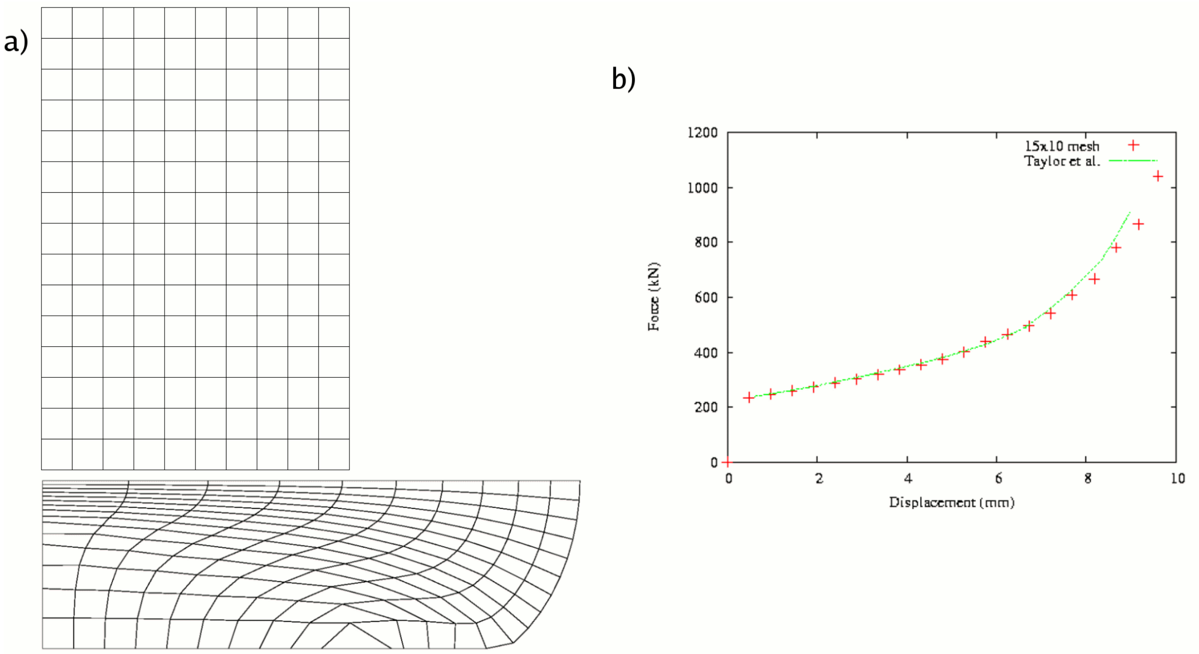

Figure 21. a) Undeformed and deformed mesh of the axisymmetric billet. b) Force-displacement curve.

Following the numerical verification, the performance of the models were tested against actual experimental data to show that the mathematical models can correctly describe the actual physical behaviour of a fuel rod. These validation cases were run with FRAPCON and FRAPTRAN now applied with the created finite element models. An example of one of the FRAPCON validation cases, an inpile creep experiment IFA-585 conducted in Halden [6], is shown in Figure 22. Figure 23 shows a ballooned region in a FRAPTRAN analysis.

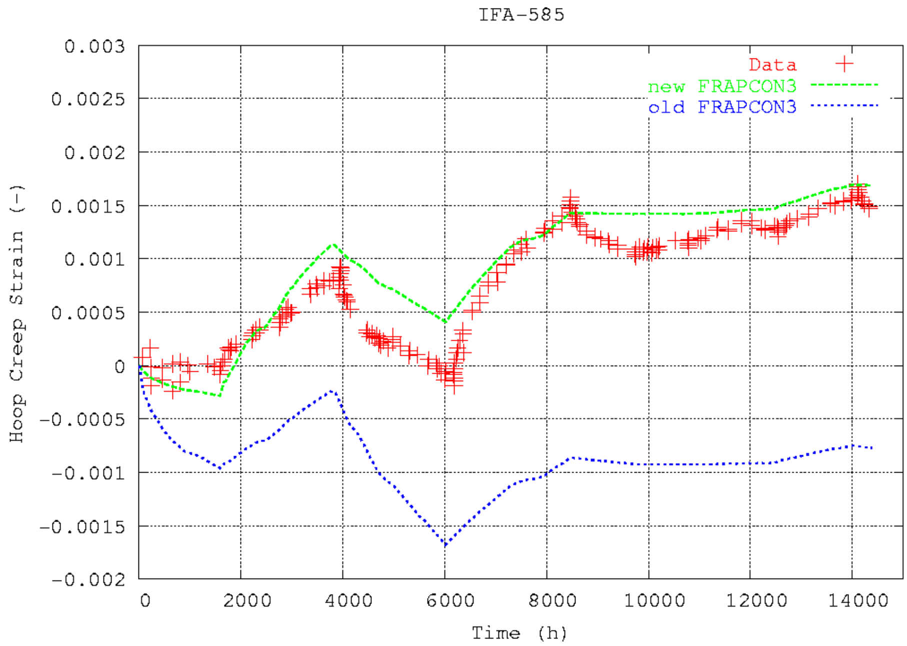

Figure 22. Hoop strain during the creep experiment IFA-585.

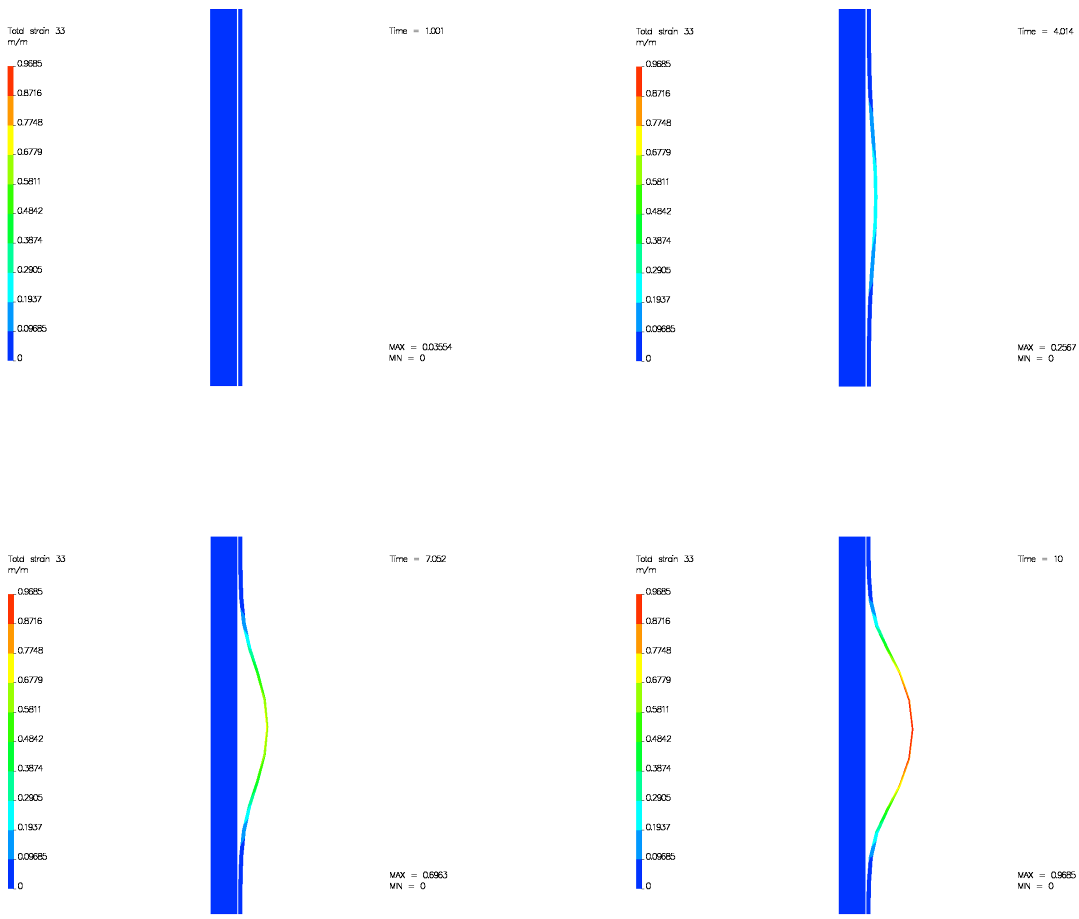

Figure 23. Hoop strains in ballooned region in the cladding.

Conclusions

A flexible, modern stress-strain analysis tool has been created for the fuel performance codes FRAPCON and FRAPTRAN. The FE model can be extended quite easily, for example the stress-strain analysis also in the fuel pellet stack is foreseen as a future development. Another interesting aspect would then be the modelling of the relocating fuel pellet fragments in the ballooned region of the fuel rod.

Although the implemented FE model is quite extensive, it has some limitations. For example, the model assumes isotropic material behaviour and the extension of modelling to the large strain anisotropy is not at all trivial matter. The mechanical behaviour of unirradiated Zirconium alloy cladding is anisotropic. However, irradiation tends to decrease anisotropy and the anisotropy effect in the mechanical properties of high burnup cladding is not significant. Even with its current-time limitations the implemented new approach of mechanical modelling means an extensive elaboration and a significant improvement over the old simple thin shell modelling.

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4. Integrity and life time of reactor circuits (INTELI)

4.1 INTELI summary report

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Abstract

The research in four new technological areas aiming to the improvement of structural safety of mechanical components and pipings is presented. These are primarily applied to primary piping but the same approach can be applied to other components as well. Risk informed inservice inspection is a new concept that can be applied to all safety relevant systems of the plant. Similarily the computer simulation of ultrasonic inspection can be applied to all components where ultrasonic inspection is used to evaluate the structural integrity and high reliability of inspection is expected. Besides these two practical approaches the research on corrosion – water chemistry interaction and fluid – structure interaction is summarised.

Introduction

The integrity of the reactor circuit is of outmost importance both in the reactors being currently in operation and in the possible new reactors. The experience gained in operation of existing plants has revealed many loading mechanisms and ageing phenomena that have not been taken into account in the design of the plant. In order to be able to simulate reliably the behaviour of the piping during operation the loading conditions, material and geometrical data as well as the environmental factors have to be known and modelled with sufficient accuracy. In a pressurized component the possibility of fracture has to be considered in the design and it has to be verified that a crack in the wall can be detected in time and the necessary corrective actions can be started.

In INTELI-project several new approaches and technologies to improve the structural safety of mechanical components and pipings are studied. From these four selected areas of research are presented here and the main results achieved in these areas during the first two years of SAFIR Programme are summarized. The selected areas are

- risk informed inservice inspection
- computer simulation of ultrasonic testing
- fluid-structure interaction analysis
- corrosion – water chemistry interaction.

Risk informed inservice inspection

Risk-informed in-service inspection (RI-ISI) applications have become increasingly attractive due to the possibility to re-direct the inspection efforts in an optimal way, leading to both an increase in safety and a decrease in inspection costs. In RI-ISI approach, the aim is to redefine the inspection programme taking into account the results of the probabilistic safety assessment, PSA. The basic idea is to reduce inspection activities in locations with low risks and identify the riskier locations where the inspection efforts should be concentrated. The main inputs for the decision making are the estimates of the consequences derived from the probabilistic safety assessments (PSA) and piping leakage and rupture potential.

In a quantitative RI-ISI approach, the degradation potential of piping should be quantified in probabilistic terms. This means that the frequency of leaks and breaks should be estimated. However, there are several degradation mechanisms that are not covered with the available tools, and also the structural reliability models may result to quite different results. Operating experience is scarce, which limits the use of statistical approaches.

There are basically the following possibilities to estimate the leak and rupture frequencies of piping:

- use of probabilistic fracture mechanics / structural reliability models
- statistical estimation from experience data, e.g. large databases
- use of formal expert judgement based on e.g. deterministic structural models.

The different possibilities were tested with a numerical example that was based on data representing a BWR weld with IGSCC as the expected degradation mechanism. In this specific case the numerical results varied between 1E-05–4E-04 for yearly leak frequency and 1E-08–6E-07 for yearly rupture frequency. It should be noted that inspections have an important effect on the leak and break estimates.

It is strongly recommended to use structured expert judgement especially in cases where there are no validated structural reliability models available. For RI-ISI purposes, it should be sufficient to assign an order of magnitude for leak and break frequencies. As the rupture frequencies to be evaluated are very small, the assignment of a probability is very difficult. This could be helped by using quantitative results of some other cases and ask experts their opinion in relation to these [1].

The amount of needed input data for probabilistic piping degradation analyses is considerable, and it is important that the data is reliable and up-to-date. In order to facilitate the piping reliability analyses for e.g. RI-ISI applications, the accessibility of plant specific data should be improved.

Computer simulation of ultrasonic testing

The applicability of a computer program developed for simulation and modelling of ultrasonic testing has been studied. The aim of the current work is to assess the possibilities to apply computer program to simulate the real inspection to be performed at site. By applying simulation at least a part of experimental measurements necessary for the qualification of the inspection system could be avoided. Normally, the inspection qualification is made by using several test blocks representing the real inspection object and the possible defects in it. In case of inspection objects having difficult geometry the production of test blocks containing intended defects is very expensive and remarkable savings could be achieved by applying computer simulation. The performance of different inspection systems can be assessed by varying the inspection approach and the defect parameters, locations, orientations etc. in computer modelling.

The versatile program modules of CIVA can be applied to model sound field produced by an ultrasonic transducer and to simulate the response and indications expected during the ultrasonic inspection.

By the CIVA_US -program the ultrasonic sound field created in the material when the transducer is moving along the surface can be simulated. Figure 24 is showing simulations of sound fields when the transducer is moving along a rough surface. The irregularities of the surface cause remarkable changes in the shape of the sound field and in the distribution of sound pressure within the beam.

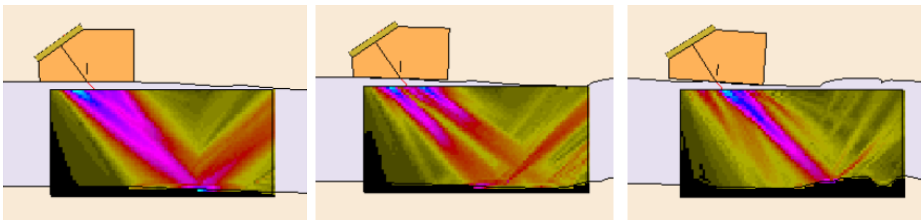

Figure 24. Simulation of sound field of the ultrasonic transducer moving along a rough surface. Sound pressure distribution presented in colour.

Another example of simulation is shown in Figure 25. In this case the ultrasonic inspection of a 40 mm thick plate containing a weld and two reference reflectors (side drilled hole Ø2 mm and planar defect 10 x 10 mm at weld root) has been simulated. In simulation the real geometry of the weld (weld root, counterbore) have been taken into account [2].

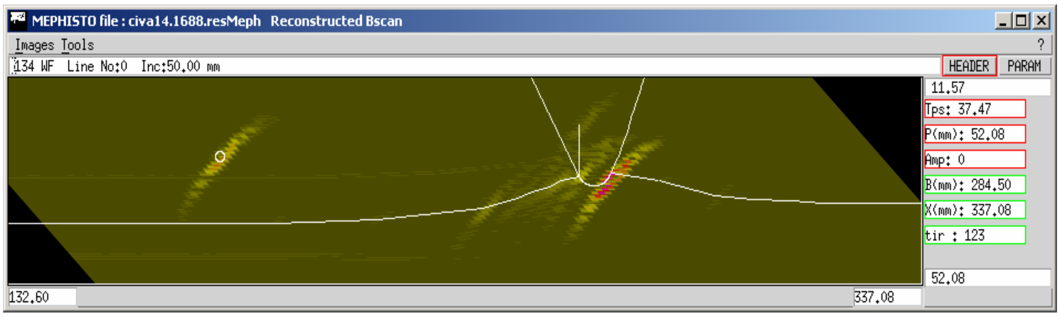

Figure 25. Simulation of the ultrasonic testing of a plate containing a weld. The transducer simulated is a shear wave transducer 45°, 2 MHz. Both the signals measured from the reference reflectors (hole and planar defect) and from the geometry of the back wall have been simulated.

The simulations shown in Figures 24 and 25 demonstrate the basic capabilities of the CIVA-software. From these relatively simple simulations the work continues in 2005–2006 to more challenging simulations where the geometry of the inspection object is more difficult, e.g. a nozzle-to-pipe weld and the position of the postulated defect more difficult from the inspection point of view.

Fluid-structure interaction analysis

Fluid-structure interactions are investigated for the POOLEX tests, where air and steam is injected into a water pool. The work consists of three subtasks. First, a program using method of images has been implemented and used in first test calculations of condensation induced water hammer. Second, first version of homogeneous two-phase model for water and steam has been implemented in the CFD code for modelling the pool experiments. Third, the coupling of the CFD and structural analysis codes is being improved by taking into use bi-directional fluid-structure interaction package ES-FSI, which is suitable for modelling linear elastic deformations of structures.

Method of images

The primary goal of the subtask was to find analytical or numerical methods easier than direct three-dimensional fluid flow calculation to estimate pressure loads onto a BWR suppression pool during discharge of steam or air or during chugging phenomenon. The Method Of Images (MOI) was chosen because of its relative simplicity and robustness. MOI is a technique for solving the Poisson equation for incompressible potential flow:

$$\nabla^2 p(x,y,z) = -\frac{4\pi S(x,y,z)}{r(x,y,z)}, \text{where}$$

$$r = \left(x^2 + y^2 + z^2\right)^{1/2}$$

The boundary conditions are disappearing pressure gradient in the normal direction on the walls and zero pressure at water surface.

A Fortran program has been implemented for solving the pressure and first test runs have been performed for the POOLEX pool. An important task in this method is choosing the source terms for the Poisson equation, which should be investigated in more detail.

Homogeneous two-phase fluid model for Star-CD

In order to simulate evaporation/condensation phenomena, a homogeneous two-phase fluid model has been implemented in the commercial Star-CD computational fluid dynamics code. In the model, all material properties, such as void fraction, density, temperature, viscosity etc., are defined as functions of fluid pressure and enthalpy. There is no transport equation for the void fraction because it is determined by pressure and enthalpy. This way all two-phase problematic matters have been included in the material property functions. In consequence, Star-CD has to solve just a single-phase fluid flow problem. The fluid-wall heat transfer has been modelled using standard models found in the literature.

The homogeneous two-phase fluid model has two difficult properties. Firstly, the temperature – enthalpy relation cannot be inverted in the two-phase region. This causes problems when standard energy equation is in use because the energy equation relies on the $h = c_pT$ relation. The problem has been circumvented by defining a new energy equation where the energy flux is determined in terms of energy gradient. Secondly, the fluid density can vary from pure steam (1 kg/m^3) to pure water (1000 kg/m^3) in the flow domain. This variation depends on pressure. This means that the momentum, continuity and energy equations depend strongly on pressure via density.

As a test case a system where superheated steam in steel pipe enters a vessel filled with cold water was simulated. The system represents the POOLEX test facility and the experiments carried out there. Figure 26 shows temperature and void fraction obtained in a steady-state simulation. Here, the pressure and temperature fields agree quite well with measurements. Also the water level in the pipe agrees well with the measurements.

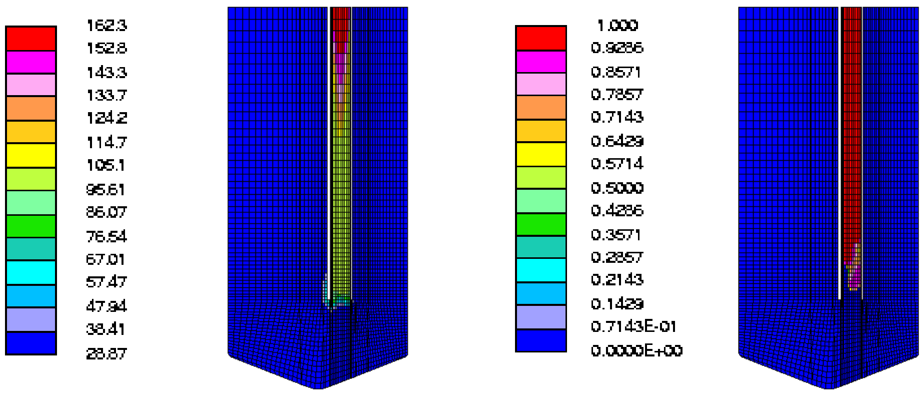

Figure 26. Calculated temperature (left) and void fraction (right) in the POOLEX test facility during blowdown of steam.

Corrosion – water chemistry interaction

The subproject "Corrosion-water chemistry interaction" of the INTELI/INPUT-project has three main focuses: 1) the modification of the high temperature titration equipment developed in earlier FINNUS-program to reach relevant temperature (288^{o}C) and to maintain constant solution composition in studying the adsorption and surface complexation of species at the oxide film/coolant -interface, 2) to study the effect of strain and strain rate on the behaviour of oxide film and on the initiation of stress corrosion cracking on AISI 316L NG, 3) the development of the channel flow electrode technique to study the effect of hydrodynamics on the dissolution of species and the protective properties of oxide films at different temperatures, pressures and surfaces.

Modification of the high temperature titration equipment

The construction materials used in coolant systems in nuclear power plants become covered with oxide films as a result of exposure to the aqueous coolant. As a result of the contact between the material surface and the coolant containing highly energetic long-lived active particles, radiation field build-up occurs in the out-of-core components and systems. In order to evaluate the rate of the incorporation of active particles into oxide films on material surfaces adsorption studies are needed both at ambient and at high temperatures. To be able to calculate the rates of adsorption of different ions (e.g. Zn, Ni, Co) to the material surfaces, experimental data gained at the relevant conditions are needed for reliable estimation of the surface reaction rates and equilibrium constants. The development of experimental measurement techniques to evaluate the surface kinetics and incorporation of detrimental cationic species to material surfaces was started in former FINNUS-programme and it has been further continued in INTELI-project.

The experimental approach has been modified so that the actual process temperature (288^oC in Boiling Water Reactors (BWR)) and pressure can now be obtained. No extrapolation of low-temperature data concerning the equilibrium constants of the reactions is needed. The high temperature pH electrodes with NiO/Ni reference pair developed at VTT since 1980's have been modified so that the activation of the electrode configuration becomes better. Better activation has been obtained mainly by pre-oxidation of the NiO/Ni powder in oven or by using Cu_2O/Cu reference pair in electrodes [3].

Effect of strain and strain rate on the behaviour of oxide film and on the initiation of stress corrosion cracking on AISI 316L NG

Environmentally-assisted cracking (EAC) can be understood as a localized deformation process accelerated by local corrosion in addition to mechanical stresses or strains. The cracking of structural materials in nuclear power plants may proceed along grain boundaries, i.e. intergranularly (IGSCC), underlining the role of dissolution, or through the grains, i.e. transgranularly (TGSCC), underlining the role of mechanical loading. As a possible approach to predict the susceptibility of structural materials to SCC, we have focused our attention on the role of oxide films formed on the structural materials. We have adopted the hypothesis that the rates of generation and transport of defects through the oxide films may be affected by the stress and strain applied to the bulk material. The straining of the bulk material is thought to affect the transport of ionic defects to the oxide film / metal interface resulting in vacancy flux into the bulk metal. This may determine how the vacancies react with dislocations at the interface and in the bulk metal.

In the INTELI-project work the effect of strain (ε = 0...30%) and strain rate ($\frac{\partial \varepsilon}{\partial t} = 10^{-7}...10^{-5}$) on the behaviour of oxide film and on the initiation of stress corrosion cracking on AISI 316L NG has been studied in a test series in BWR conditions (288^oC, 1 ppm O_2). The thickness and composition of the surface oxide films were analyzed for each strain level. A correlation between the content of trivalent state of chromium in the films and the electric resistance of the film has been suggested to exist based on the results [4, 5, 6, 7].

Conclusions

The basic approaches to estimate the leak and rupture frequencies of piping have been tested with a numerical example that was based on data representing a BWR weld with IGSCC as the expected degradation mechanism. For RI-ISI purposes, it should be sufficient to assign an order of magnitude for leak and break frequencies. As the rupture frequencies to be evaluated are very small, the assignment of a probability is very difficult.

By computer modelling and simulation of ultrasonic testing the performance of inspection approach can be verified without manufacturing sets of expensive test blocks containing realistic reference defect. Especially for the inspection qualification computer modelling is a valuable tool that can be applied both to optimize the inspection procedure and to demonstrate its performance.

The cracking of structural materials in nuclear power plants may proceed along grain boundaries or through the grains. As a possible approach to predict the susceptibility of structural materials to stress corrosion cracking the attention has been focussed on the role of oxide films formed on the structural materials. It is assumed that the rates of generation and transport of defects through the oxide films may be affected by the stress and strain applied to the bulk material. The straining of the bulk material is thought to affect the transport of ionic defects to the oxide film / metal interface resulting in vacancy flux into the bulk metal. This may determine how the vacancies react with dislocations at the interface and in the bulk metal.

In order to simulate evaporation/condensation phenomena in fluid – structure interaction, a homogeneous two-phase fluid model has been implemented in the commercial Star-CD computational fluid dynamics code. As a test case a system where superheated steam in steel pipe enters a vessel filled with cold water has been simulated. The system represents the POOLEX test facility and the experiments carried out there. The pressure and temperature fields agree quite well with measurements. Also the water level in the pipe agrees well with the measurements.

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4.2 Constraint corrected fracture mechanics in structural integrity assessment

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Abstract

Specimen size, crack depth and loading conditions may affect the materials fracture toughness. In order to safeguard against these geometry effects, fracture toughness testing standards prescribe the use of highly constrained deep cracked bend specimens having a sufficient size to guarantee conservative fracture toughness values. One of the more advanced testing standards, for brittle fracture, is the Master Curve standard ASTM E1921, which is based on technology developed at VTT Industrial Systems. When applied to a structure with low constraint geometry, the standard fracture toughness estimates may lead to strongly over-conservative estimate of structural performance. In some cases this may lead to unnecessary repairs or even to an early "retirement" of the structure. In the case of brittle fracture, essentially three different methods to quantify constraint have been proposed, J-small scale yielding correction (SSYC), Q-parameter and the T_{stress}.

Introduction

The use of fracture mechanics in design and failure assessment is to some extent impeded by the difficulties of quantifying the structural constraint as a function of geometry and loading. It is well known that specimen size, crack depth and loading conditions may affect the materials fracture toughness. In order to safeguard against these geometry effects, fracture toughness testing standards prescribe the use of highly constrained deep cracked bend specimens having a sufficient size to guarantee conservative fracture toughness values. An example of one of the more advanced testing standards, providing a method to determine such a "base line" fracture toughness characterization for brittle fracture, is the so called Master Curve standard ASTM E1921, which is based on technology developed at VTT Industrial Systems during the last 15 years. These "base line" toughness values have one weakness. When applied to a structure having low constraint geometry, the standard fracture toughness estimates may lead to strongly over-conservative estimates. In some cases this may lead to unnecessary repairs or even to an early "retirement" of the structure.

Current investigation applies the Master Curve method in conjunction with the T-stress correction of the reference temperature and a modified Beremin model to an assessment of a three-dimensional pressure vessel nozzle in a spherical vessel end. The material information for the study is extracted from the 'Euro-Curve' ductile to brittle transition region fracture toughness round robin test program. The experimental results are used to determine the Master Curve reference temperature and calibrate local approach parameters. The values are then used to determine the cumulative failure probability of cleavage crack initiation in the model structure. The results illustrate that the Master Curve results with the constraint correction are to some extent more conservative than the results attained using local approach. The used methodologies support each other and indicate that with the applied local approach and Master Curve procedures reliable estimates of structural integrity can be attained for complex material behavior and structural geometries.

Master Curve Analysis

The fracture toughness dataset used in the current context is a part of the 'Euro' fracture toughness dataset, where a larger number of C(T) and SEN(B) specimens were tested of a nuclear grade pressure vessel forging 22NiMoCr37 (A508 Cl.2). The test were performed for a range of C(T) specimen sizes, and in addition, a number of precracked Charpy size specimens were used to determine ductile to brittle fracture toughness transition. The data pertaining to Charpy size specimens is applied within the context of the current work, details can be found from [1].

A T-stress correction to the Master Curve reference temperature has been proposed and validated by [2]. The T-stress correction was formulated on the basis of a large experimental dataset and has been found to be an amenable tool in usage of the Master Curve method in structural integrity evaluation. The T-stress correction is performed directly to the Master Curve reference temperature, Eq. (1)

$$T_0 \approx T_0^{deep} + T_{stress}/10 \text{ for } T_{stress} \leq 0 , \tag{1}$$

where T_0^{deep} is the reference temperature value under high constraint conditions. By use of Eq. (1) the Master Curve methodology can be extended to low constraint conditions and all other aspects can be fully utilized.

Master Curve Analysis Results

The results of the multitemperature Master Curve analysis for the fracture toughness dataset is presented in Figure 27 with respect to the reference temperature. A T_0 value of -97°C is attained.

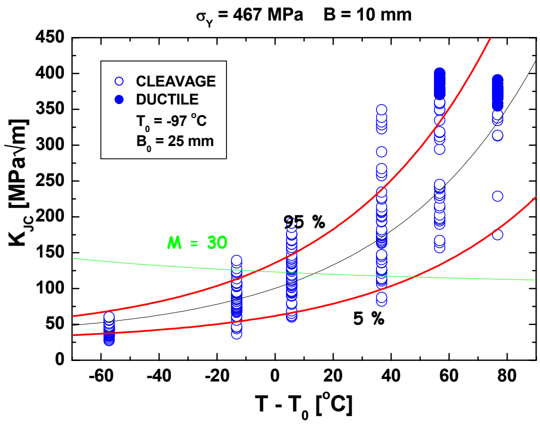

Figure 27. Multitemperature Master Curve analysis results.

Modified Beremin Model and Parameter Calibration

The failure probability of the used form of the Beremin model is given as [3]:

$$P_f = 1 - \exp\left[-\left(\frac{\sigma_w - \sigma_{th}}{\sigma_u - \sigma_{th}}\right)^m\right], \tag{2}$$

where σ_{th} is the threshold stress, σ_u is the scale parameter and m the shape parameter.

The calibration process consists of determining the variation of the Weibull stress with different values of the shape parameter to find the maximum of the log maximum likelihood equation by inputting the experimental fracture toughness results.

Structural Integrity Evaluation

The case structure was a semi-elliptical surface nozzle corner crack at the intersection of a spherical shell and a pipe. The crack extended to quarter thickness and had a width three times its depth. The loading can be considered to consist of a pressure transient. The near crack tip mesh of the model is presented in Figure 28.

T-stress evaluation

The T-stress was computed using the interaction integral routine as implemented in Abaqus 6.3-1, [4] and Warp3D [5].

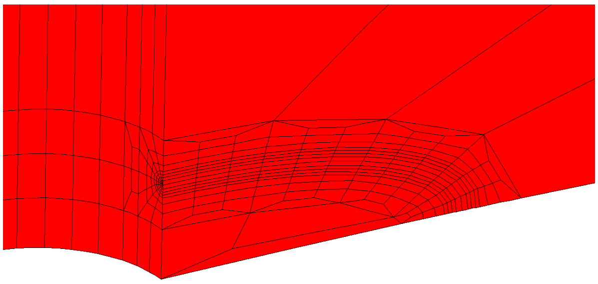

Figure 28. Near crack-tip mesh of the case structure.

Master Curve structural integrity evaluation results

The T-stress as a function of crack driving force is given in Figure 29. In the analyses, the T-stress was inferred from a length of the crack front corresponding to T-stress values not indicating drastic constraint loss as occurring near the surface of the semi-elliptical surface crack, and the same treatment was applied to the crack driving force.

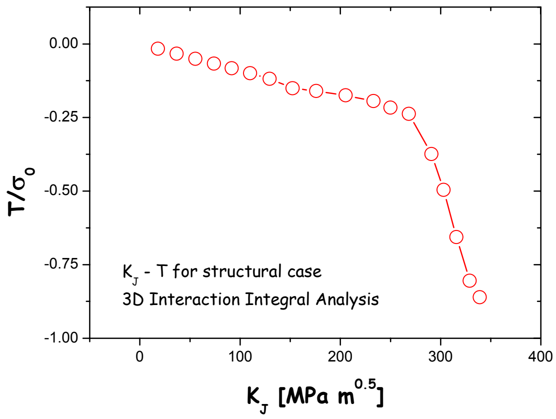

Figure 29. K_J - T trajectory used in the Master Curve analyses.

Figure 30 presents the cumulative failure probability as a function of crack driving force as attained from the Master Curve analysis. The results are given for three nominal temperatures all pertaining to the upper parts of the ductile to brittle transition region. The temperatures do not have explicit significance since they are primarily applied in comparing the two different approaches. The results of Figure 30 have a feature unusual to failure probability plots utilizing the Master Curve probability distribution. This is the decrease of cumulative fracture probability as a function of K_J after a certain value, which is approximately 250 MPa $m^{0.5}$. This results from the T-stress constraint correction, i.e. after a given value of crack driving force, which in this case is not really affected by temperature due to the small material property variations within this narrow temperature band, the structure exhibits loss of constraint, and the nominally monotonic probability distribution function attains a decreasing slope. The current K_J – T-stress trajectory results to a continuous and abrupt loss of constraint after a given value of crack driving force, resulting to an immediate loss of constraint, increase of normalization fracture toughness and decrease of failure probability.

In Figure 30 for the temperatures 20°C and -20°C horizontal lines have been added to describe the maximum values of failure probability. Since the constraint correction is made directly to the probability distribution, it is possible that it attains a decreasing value, which is in contradiction with its fundamental properties as a cumulative distribution function. As such, after the peak value the curves are no longer valid, but a constant value of failure probability corresponding to the peak value ought to be considered valid. One could state that in evaluating the failure probability in constraint varying conditions performing the constraint correction directly to the probability density function would produce a continuous and monotonic framework.

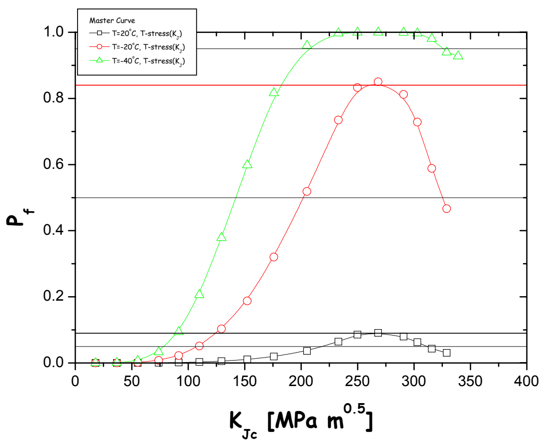

Figure 30. Master Curve analysis results for failure probability.

Local approach structural integrity evaluation results

The results attained using the modified Beremin model are presented in Figures 31 and 32. Along with the local approach results, the analysis findings of the Master Curve method are given for comparison. The Master Curve results are presented for constraint corrected and uncorrected analyses, respectively. The local approach evaluation results are provided for two calibrated sets of Weibull shape and scale parameters. The calibration results were fitted with respect to temperature and the findings used to produce the parameters used in the integrity evaluation case.

Overall the results illustrate that the Master Curve method is conservative in comparison to the local approach analyses. This feature is retained even though the loss of constraint as modeled by the T-stress is accounted for, and as such the formulations and assumptions made in derivation of the constraint modified Master Curve method are supported, since the local approach analysis naturally inherently incorporates linear-elastic and elastic-plastic constraint phenomena, within the bounds and specifications of the probability distribution function and its derivation, respectively.

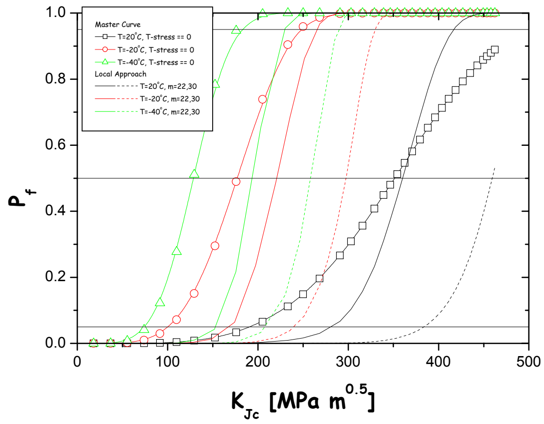

Figure 31. Local approach and T-stress == 0 Master Curve analysis results.

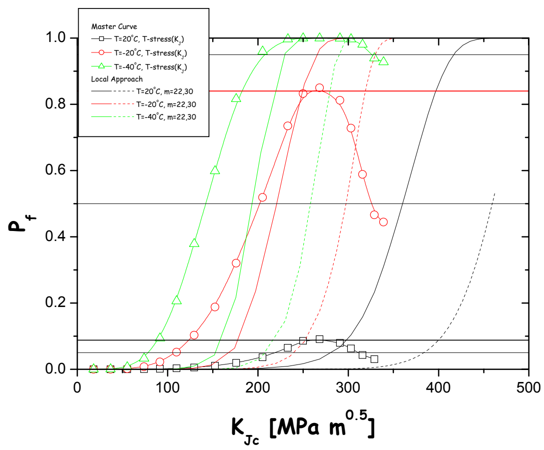

Figure 32. Local approach and T-stress (K_J) Master Curve analysis results.

The Master Curve constraint correction is seen to affect the failure probability for results attained at 20°C quite significantly, i.e. the peak failure probability is below 0.1 in the constraint corrected analysis. This can be interpreted to relate to the fact that at such a high temperature the steepness of the Master Curve overruns the crack driving force, as such leading to a drastic decrease in failure probability. In a way, one might state that for this temperature the Master Curve is unconservative in comparison to the local approach estimate, but since the local approach method lacks inherent temperature dependency the model still predicts crack initiation by cleavage. On the basis of experimental results it can be argued that the Master Curve result is sound for cleavage crack initiation, or additional cleavage data ought to be used to estimate the local approach model parameters at 20°C.

Comparison of local approach and Master Curve analysis results

Figure 33 presents a number of fracture toughness values attained with the different analysis methods corresponding to different failure probability levels. The datapoints have been selected from failure probabilities of 0.05, 0.5, and 0.95, respectively.

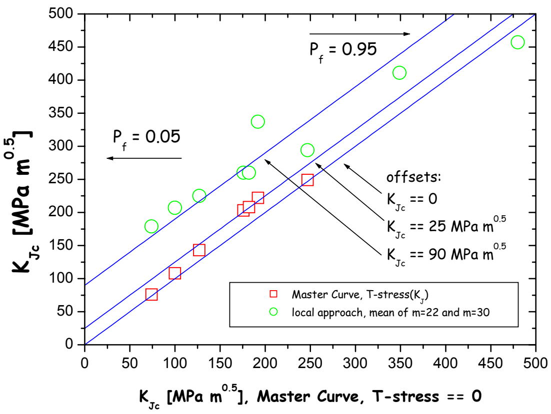

Figure 33. Comparison of Master Curve, constraint corrected Master Curve and local approach analysis results.

Comparison of results of Figure 33 indicates, by making a crude assessment on the differences of the different approaches, that the two different Master Curves differ around 25 MPa $m^{0.5}$ and the local approach estimates differ from the unmodified Master Curve by approximately 90 MPa $m^{0.5}$. Considering median fracture toughness values, the relative differences are of the order of 10–20% for the Master Curves and 40% when comparing the local approach results to the unmodified Master Curve.

Summary and conclusions

Master Curve analysis and calibration of local approach parameters for a modified Beremin model were performed for a 22NiCrMo37 steel in the ductile to brittle transition region. The results were utilized in carrying out a study to investigate the fracture behavior of a case structure. The results of the work can be concluded as follows:

- The constraint corrected Master Curve method produced results that were conservative compared to local approach estimates, the differences ranging up to 40%. The Master Curve results reflect the fact that the applied constraint correction is based on the elastic T-stress.

- The quality of the Master Curve evaluation results is most affected by the process used to evaluate the T-stress, and care must be taken when applying the T-stress under situations dangling near its validity bounds.

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4.3 Fatigue behaviour of 316 NG stainless steel in EPR primary circuit conditions

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Abstract

Strain controlled constant amplitude and variable amplitude fatigue tests were performed for 316 NG stainless steel in PWR water 320°C. The environment reduced fatigue life as predicted. Variable amplitude test results indicate a further reduction in life by a factor of 5, which is greater than expected. Variable amplitude straining in EPR environment resulted to specimen failure before the fatigue usage factor according to the ASME III design curve reached unity. In other words, the ASME III design curve was conservative for constant amplitude, but non-conservative for variable amplitude straining in environment. Critical tests to confirm or replace this tentative observation are recommended.

Introduction

The design by analysis procedures introduced in ASME Boiler and Pressure Vessel Code Section III [1] aim to prevent formation of fatigue cracks during the specified service period. Significant fatigue cycles are often caused by thermal loads. They occur in small numbers, but result to notable plastic strains.

The effect of high temperature water environment on fatigue damage rate became an issue of particular interest when U.S. NRC was preparing a position in regard to first plant license renewals for the American nuclear industry in 1991. Studies performed in Argonne National Laboratory (ANL), USA [2–5] and in Japan [6–8] indicated remarkable environmental effects, Figure 34.

Fatigue design curves given in the ASME Code Section III are based on strain controlled low cycle fatigue tests in room temperature. Although it was stated that "protection against environmental conditions such as corrosion and radiation effects are the responsibility of the designer", the margin (a factor of 20 on cycles) between the mean curve and design curve is supposed to cover some environmental effects. [9]

The factor of 20 was broken down into 2 for scatter, 2.5 for size and 4 for environment, surface effect and so forth, but this breakdown seems to have been highly speculative and potentially unrealistic. The breakdown was not officially published in the code or its Criteria document [9], but the documentation gives reason to assume that the designer should either verify that the fatigue life reduction due to environment is less than 4 for the selected material environment combination, or take additional measures to cover the environmental effects. Additional measures could be, e.g., use of alternative design curve based on relevant material testing, or adoption of specific environmental correction factors according to an F_{en} approach [10]. The latter approach has already been adopted to TENPES guidelines in Japan [11] and is being considered for ASME Code implementation. [12]

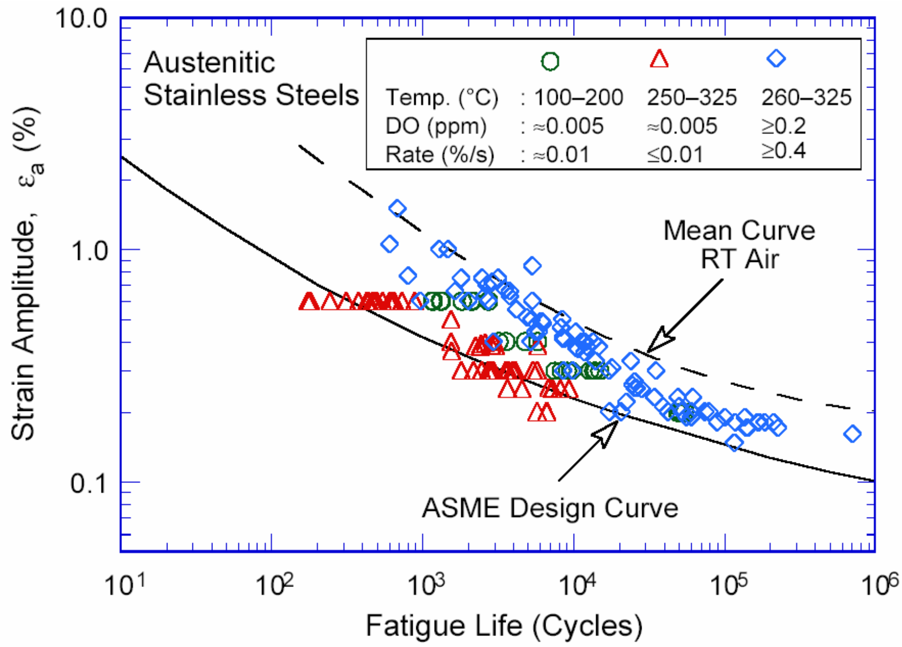

Figure 34. Grouped fatigue data for austenitic stainless steels in water. [5]

The YVL guide 3.5 issued by the Radiation and Nuclear Safety Authority (STUK) states that influence of environment shall be accounted for in fatigue assessment. Quantitative requirements are beyond the scope of the YVL guide, but it lists the main environmental factors influencing fatigue life of materials used in primary circuits and specifically states that justification is required, if ASME III fatigue curves are used for assessing environment assisted fatigue. [13]

Tests compatible with the ASME III Design by analysis procedure are needed also in LWR environments to fulfill the requirements of YVL 3.5 and to assess the significance of different environmental parameters for fatigue of reactor components. Furthermore, it has been shown that mechanisms of environmental effects are not the same for stainless steels as those proposed for carbon steels, but the real mechanisms are not yet known [5]. In long run, mechanism based models should be developed for fatigue of all primary circuit materials. For these purposes, VTT has developed a novel facility based on bellows loading technology.

A few demonstrative tests were conducted to measure the combined effect of environment and variable amplitude loading. The aim was to demonstrate that the developed technology is suitable for direct strain-control for LCF, HCF and spectrum fatigue tests in hot water.

Experimental

Low cycle fatigue testing requires strict attention to load train rigidity, specimen design, and alignment. The large degree of plasticity observed in fatigue testing of stainless steels and lack of linear response in elastic range mean that strain shall be controlled in a smooth gage section. For strain measurement a LVDT is mounted co-linear with the specimen with non-backlash knife blade attachments on each side of specimen. The arrangement worked well in the prototype verification tests in 1999, but practical difficulties appeared in the autoclave tests. A modified mounting system designed in 2003 compromised the strain measurement accuracy and a third generation design was used for the here reported tests. The installation work was still difficult, but it is considered successful as

- no sliding of strain measurement was observed
- the overall quality of measured stress-strain data was convincing
- cracks initiated in the specimen midsections between the knife edges.

The design of the unit and specimen is illustrated in Figure 35. Verification of the alignment and dynamic performance of the unit has been summarised previously [14].

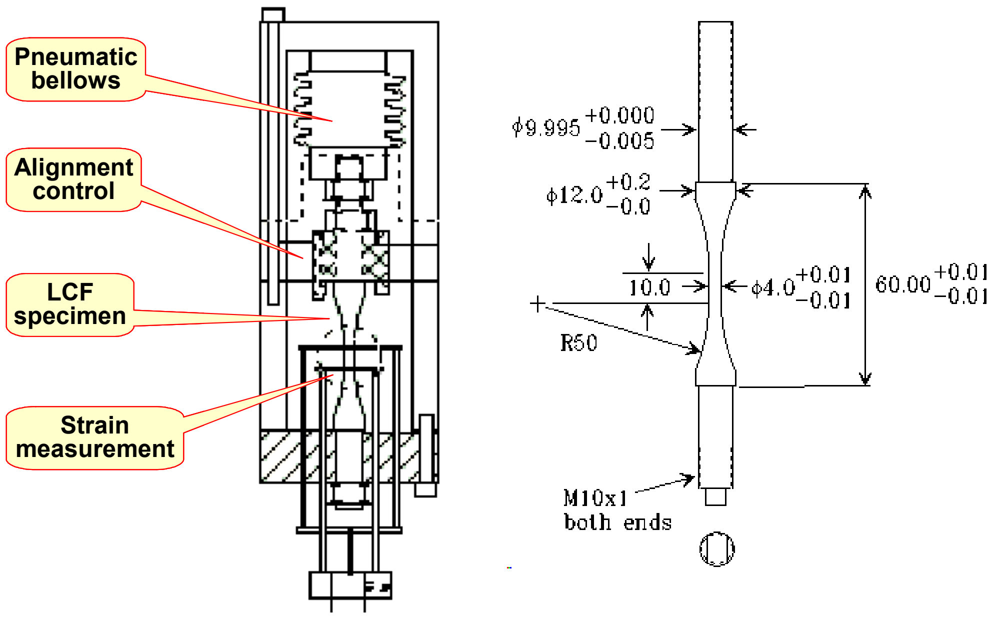

Figure 35. Schematic of the fatigue unit and specimen geometry.

Small fatigue units provide a possibility of testing multiple specimens simultaneously in the same water circulation loop, and thus, to produce high quality data for direct comparison and statistical analysis. This is particularly valuable for determining the influence of various environmental parameters. Previously four units were installed in one single autoclave, but the current set-up consists of four separate autoclaves in the same loop, Figure 36. The air tests were performed in MTS 100 kN rig with Test Star digital control unit and precision alignment collet grips.

Figure 36. Four fatigue autoclaves in a hot water circulation loop.

Test material and test environment

Chemical content (wt%) of the 316 NG stainless steel was as follows: C 0,01; Si 0,6; Mn 1,4; Cr 16,9; Ni 11,2; Mo 2,6; P 0,029; S 0,028. From the batch measured four yield and tensile strengths varied between: $R_{p0,2}$ = 246 ... 256 MPa and R_m = 562 ... 575 MPa. Round specimens with diameter of 4 mm were carefully machined according to Figure 35. The specimen surfaces were smooth, but not polished.

Environmental parameters of the low oxygen water were as follows: temperature 320°C; pressure 125 bar; pH at room temperature 5,1; dissolved oxygen : < 0,1 ppm; chlorides < 0,15 ppm; fluorides < 0,15 ppm; dissolved hydrogen 25–35 cm^3 (TPN)/kg water; boric acid 2500 ppm; lithium quantity necessary to adjust the pH at 7,0; conductivity 2–40 µmho/cm. A low flow rate water circulation was applied.

Fatigue tests were performed strain controlled with sinusoidal waveform, either with constant amplitude or by repeating a pre-defined block of 50 strain cycles of variable amplitudes, Figure 37.

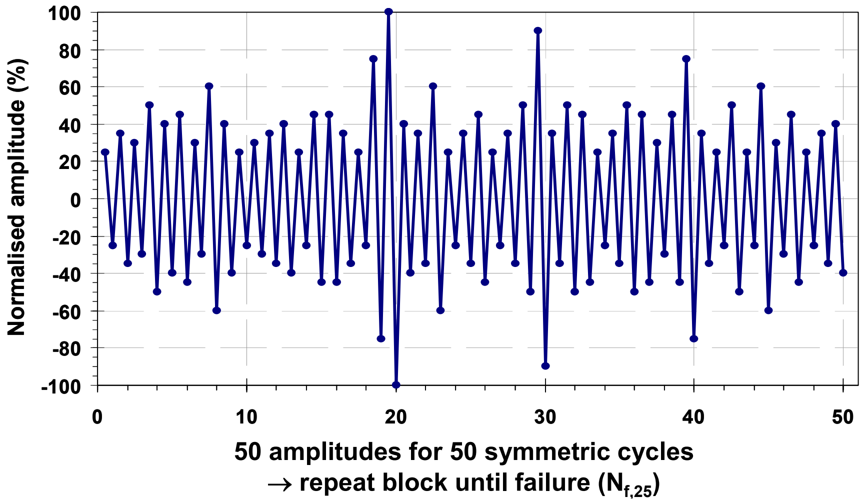

Figure 37. Randomised amplitude sequence for spectrum straining tests.

Presentation of results for spectrum straining

The methodologies used for analysing spectrum straining tests have been presented in more detail elsewhere [15–17]. In general terms, an equivalent constant amplitude loading that is representative to variable amplitude loading is sought for.

Commonly used definitions for "equivalent" loading may sometimes be misleading and their applicability is limited to certain damage equations. For nonlinear damage equations equivalent loading can only be obtained by calculating a nonlinear average of amplitudes. For this average, the amplitudes are weighted by the damage (usage factor) caused by each cycle. For corresponding number of cycles, the cycles are summed up such that cycles causing more damage than the equivalent loading are counted as more than one and the smaller ones as less than one, Eqs. 1–2.

$$\varepsilon_{a,eq} = \frac{\Sigma\,[\,n_i \cdot D_i \cdot (\varepsilon_{a,i})\,]}{\Sigma\,[\,n_i \cdot D_i\,]} \qquad N_{eq} = \Sigma\left\{n_i \cdot \left(\frac{D_i}{D_{i,(\varepsilon_{a,eq})}}\right)\right\}, \tag{1}$$

where D_i is an arbitrary damage function (in our case ASME design curve):

$$\frac{1}{N_f} = D_i = f(\varepsilon_a) \tag{2}$$

Calculation of F_{en} for environmental effects

The ASME best estimate curve was used as a reference for F_{en} predictions. Calculations of F_{en} as function of temperature (T), dissolved oxygen content (DO) and strain rate (ε') for wrought stainless steels followed the PVRC CLEE proposal [10], Eq. 3:

$$F_{en,\ nom} = \exp\ (0.935 - T^*O^*\varepsilon'^*)\ , \qquad (3)$$

where	$\varepsilon'^* = 0$	$(\varepsilon' > 0.4\%/\text{sec})$
	$\varepsilon'^* = \ln(\varepsilon'/0.4)$	$(0.0004 \leq \varepsilon' \leq 0.4\%/\text{sec})$
	$\varepsilon'^* = \ln(0.0004/0.4)$	$(\varepsilon' < 0.0004\%/\text{sec})$
	$T^* = 0.0$	$(T \leq 180°C)$
	$T^* = (T-180)/40$	$(180°C < T < 220°C)$
	$T^* = 1.0$	$(T > 220°C)$
	$O^* = 0.260$	$(DO < 0.05\ \text{ppm})$
	$O^* = 0.172$	$(DO \geq 0.05\ \text{ppm})$

Results and discussion

Cyclic deformation

The strain cycles are preset for each test, but the stress cycles depend on material response to these strain histories. Figure 38 shows an example of hysteresis loops during the final phase of a test. Figure 39 summarises stress variation during another constant amplitude test. Figure 39 also illustrates our failure criterion $N_{f,25}$ and its definition as the number of cycles to 25% drop of peak stress.

Non-symmetric flow stresses consistently generated compressive mean stresses, Figure 39. As expected, stress amplitudes are smaller in elevated temperature. The studied steel experienced initial hardening and subsequent softening, but the amount of initial hardening and subsequent softening is less prominent in environment at a lower strain amplitude (0,3%). Furthermore, both these specimens spent most of the test duration in a secondary hardening phase before the final loss of strength due to cracking, Figure 40.

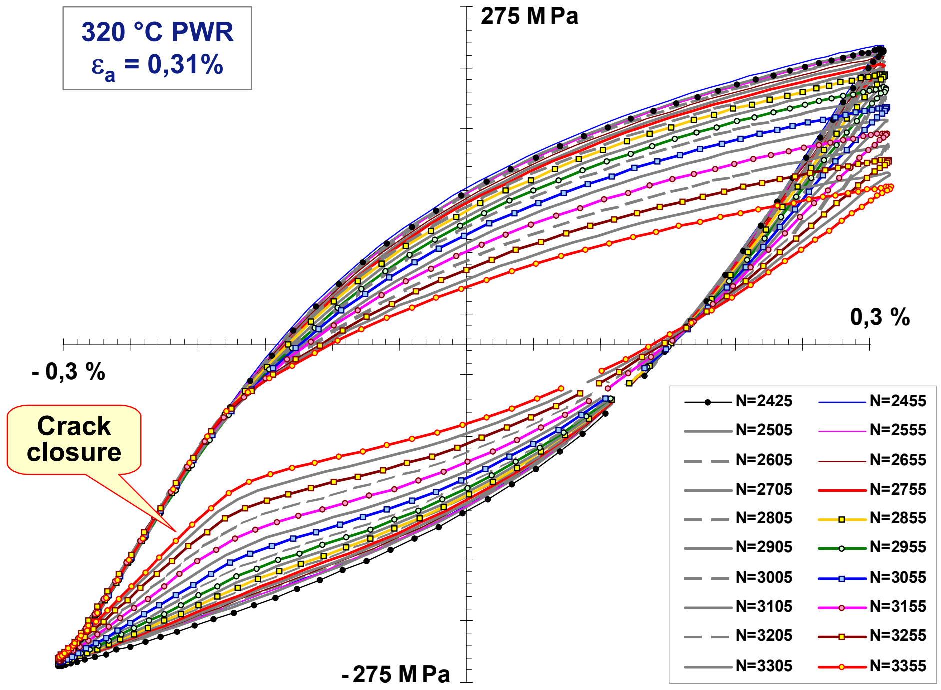

Figure 38. Measured stress – strain responses during a test in 320 °C PWR environment.

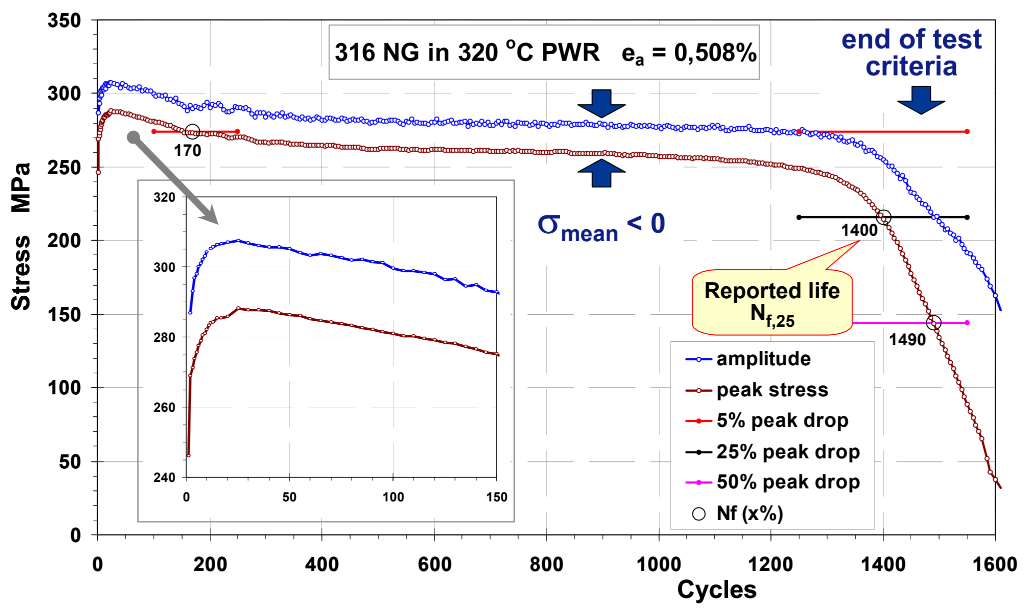

Figure 39. Measured peak stress and amplitude as function of elapsed cycles.

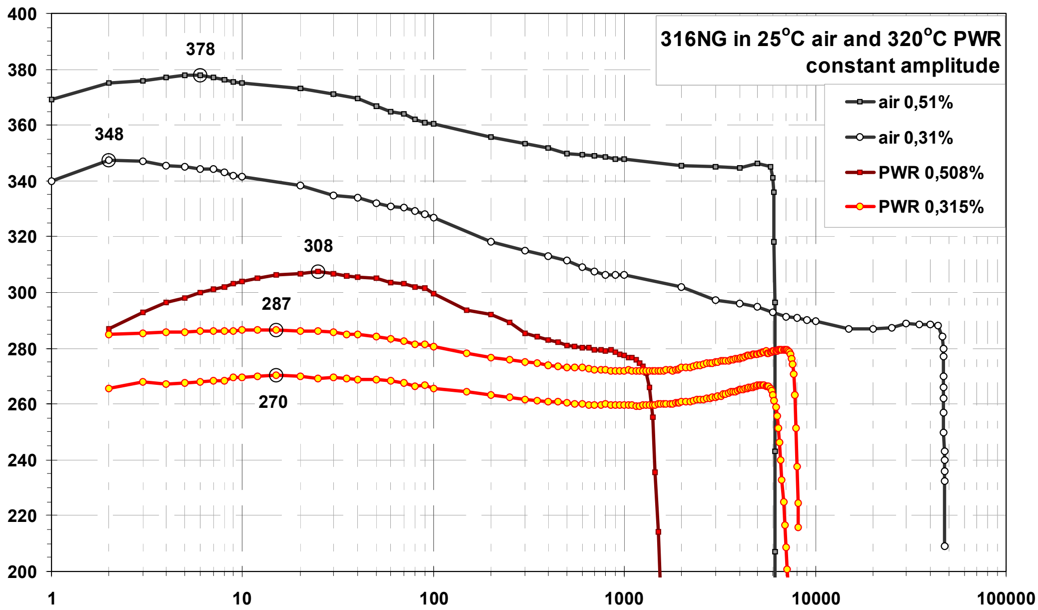

Figure 40. Measured stress amplitude in constant amplitude tests.

Fatigue lives

The fatigue test results are summarised in Table 5 and in Figure 41. A prediction of fatigue life was made for each specimen tested in environment. A nominal F_{en} factor is used to reduce the best estimate (mean life) curve, which is the basis for ASME III fatigue design curve for stainless steels [9]. These predictions are compared to the experimentally obtained fatigue lives in the two rightmost columns in Table 5 and in Figure 41. F_{en} is unity for the air tests, and increases according to the environmental effect.

Table 5. Summary of fatigue test results and comparison of measured environmental effects to predicted ones ($F_{en} = N_{f,air} / N_{f,environment}$).

environment	Freq. Hz	strain rate % / sec.	ε_a %	$N_{f,25}$ cycles	mean curve / N_f (F_{en}) measured	predicted
air 25C	0,5	1,020	0,510	6120	1,41	1,00
air 25C	0,5	0,620	0,310	47000	0,99	1,00
air 100C	0,1	0,168	0,420	15600	1,03	1,00
air 25C	spectrum	0,800	0,20…0,80	9650	2,3	1,00
PWR 320C	0,1	0,100	0,250	15600	8,65	3,65
PWR 320C	0,01	0,016	0,400	2650	7,08	5,88
PWR 320C	0,01	0,020	0,508	1400	6,29	5,53
PWR 320C	0,01	0,013	0,315	6660	6,61	6,26
PWR 320C	0,01	0,013	0,315	7800	5,64	6,26
PWR 320C	spectrum	0,039	0,11...0,48	4600	30,58	4,67
PWR 320C	spectrum	0,065	0,19...0,78	1250	20,31	4,09

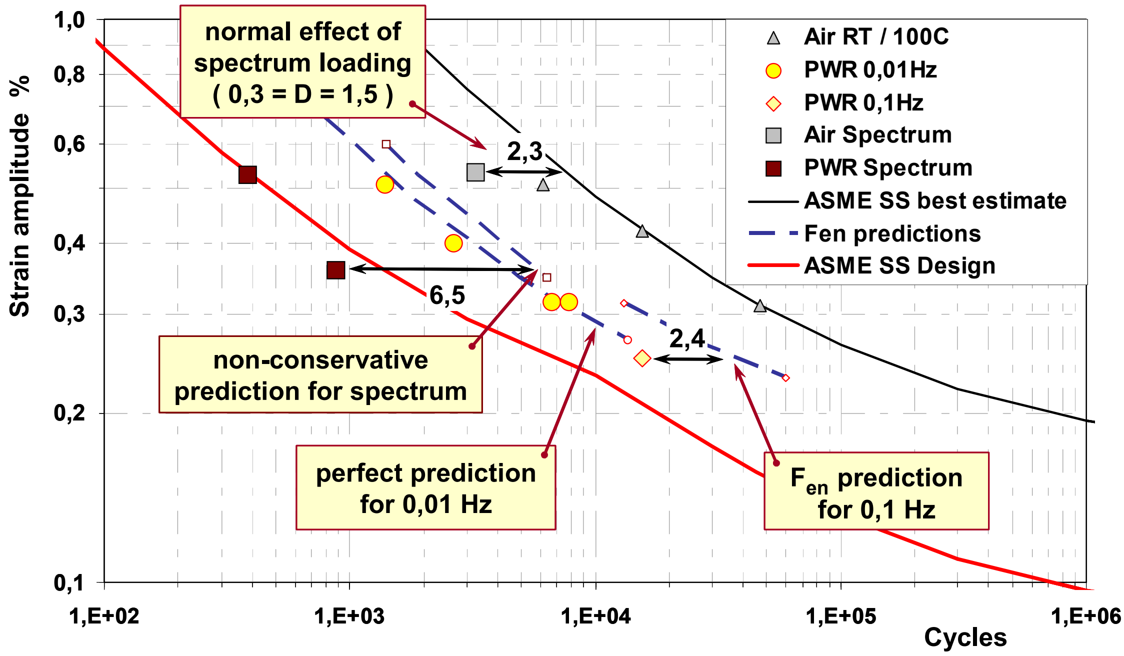

Figure 41. Summary of strain – life results and comparison to F_{en} predictions.

The F_{en} prediction for the previously tested data point at 0.1 Hz (ε_a = 0.25%) was non-conservative by a factor of 2,4. But it is a single data point and the prediction would have been better, if the Argonne mean curve were used as the reference for F_{en} prediction. On the other hand, the prediction was very good for the constant amplitude tests with 0.01 Hz frequency, Figure 41.

Effect of spectrum straining

The most striking observation is that the life reduction due to spectrum straining in environment is greater than expected, even larger than would have been predicted for summed environmental and spectrum effects. Spectrum straining with strain amplitudes ranging from 0.19% to 0.78% lasted until the fatigue usage factor according to the ASME III design curve was exactly one. The test with strain amplitudes ranging from 0.11% to 0.48% lasted until the fatigue usage factor was only 0.65.

Life reduction due to spectrum straining in air was as expected: 2.3 when compared to the best estimate curve, and even less, if comparing to the constant amplitude data point at 0.51%. Life reduction factors between 1 and 3 would be in line with our previous experience [18]. According to literature the usage factor at failure in variable amplitude tests is normally found to lie between 0.3 and 1.6. This is also the reason, why some design codes set the allowable damage to 0.3 instead of 1.

Unfortunately, there are no more spectrum straining tests performed in LWR environments, but the current results indicate that variable amplitude test results may cause a further reduction in life by a factor of 5 in environment. This opens a question on potential synergistic effect of environment and variable amplitude straining.

Cyclic Stress Strain Curve

In contrast to constant amplitude test results, Cyclic Stress Strain Curves (CSSC) measured by spectrum straining follow Hollomon strain hardening model very well, Figure 42. The deformation mechanisms of alloy 316 NG depend on strain amplitude, but we can assume that the regularly repeated largest strain cycles control the microstructure (dislocation density, number of slip bands, cell size or other characteristic dimensions), which further defines the stress strain responses and the resulting CSSC. In other words, spectrum straining tests provide CSSC's with constant microstructure. This issue is relevant also for stress analysis. The largest strain amplitudes occurring in operation define the deformation mechanisms available for smaller cycles also in real components. Consequently, CSSC's derived from constant amplitude data should be applied by care for service loading.

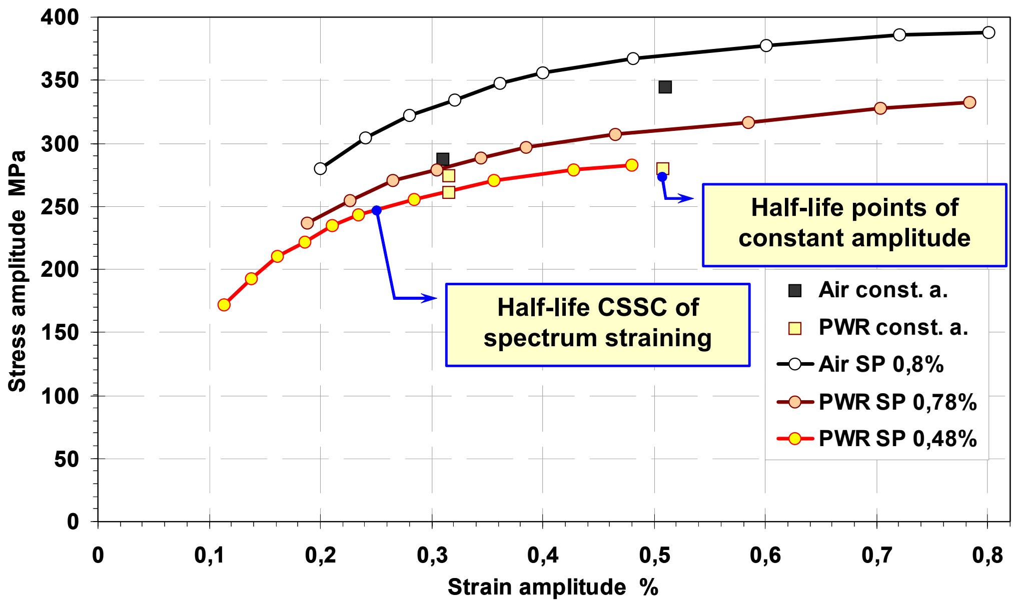

Figure 42. Measured CSSC's in air and in 320°C PWR.

Conclusions

The obtained data for 316 NG steel indicates a notable effect of PWR environment in 320°C, but the constant amplitude data lies above the ASME III design curve in good agreement with the data published by the Argonne National Laboratory and Japanese laboratories. Our results support the currently applied design rules and F_{en} approaches. In particular, constant amplitude results at 0.01 Hz sinusoidal loading comply with F_{en} predictions according to the methodology proposed to ASME Code by the PVRC.

Spectrum straining in environment led failures when the fatigue usage factors according to the ASME III design curve were 1 and 0.65. This means that variable amplitude testing caused a further reduction in life by a factor of 5 in environment and opens a question on potential synergistic effect of environment and variable amplitude strain. Critical tests to confirm or replace this tentative observation are recommended.

The new facility is ready for project use. It provides advantages which are expected to reduce the scatter in experimental data.

The cyclic deformation mechanisms were monitored in terms of cyclic hardening and softening, hysteresis loops and CSSC's. It was concluded that the obtained data is suitable for clarifying the currently unknown mechanism(s) of environmental effect in LCF for stainless steel and for explaining the influence of critical environmental and loading parameters, but more data should be gathered before the model development can be successful.

Acknowledgements

The work reported is part of the Finnish national research programme on Safety of nuclear power plants in 2003–2006. The encouragement and discussions within the expert support group of INTELI project are gratefully acknowledged. Dr. Pekka Moilanen has developed the bellows loading technology for this and many other applications. Prof. Gary Marquis, currently at Lappeenranta university of technology, was responsible of the prototype fatigue unit design and verification. Ms. Päivi Karjalainen-Roikonen has supervised the design and build of the current testing facility with four autoclaves.

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5. LWR oxide model for improved understanding of activity build-up and corrosion phenomena (LWROXI)

5.1 Kinetics of the growth and restructuring of the oxide layer on stainless steel in a light water reactor coolant

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VTT Industrial Systems

Abstract

The oxide films formed on AISI 316L(NG) in the temperature range 150–300°C in a simulated light water reactor coolant have been characterised by impedance spectroscopy and ex-situ analysis using Auger electron spectroscopy. The results show that the nature of the barrier layer does not change drastically with temperature, although the growth mechanism of the oxide film is different at 150…300°C than at room temperature. The ability of the Mixed-Conduction Model for passive films to reproduce the experimental impedance data has been tested. A procedure for the calculation of the kinetic constants of the interfacial reactions, as well as the diffusion coefficients of ionic / electronic defects in the oxide has been developed. The effect of temperature on the parameters has been quantified, and their relevance for the corrosion behaviour of stainless steel in a high-temperature electrolyte is discussed.

Introduction

All materials used in the nuclear power plants rely on a passivating oxide film. This means that the oxide film formed from the corrosion attack of water on the material will protect the underlying material against further attack. The material in itself would, according to the thermodynamics, decompose into a mixture of metal oxides without this protective layer. This deterioration of the material is also observed if the passivating film is destroyed for a certain reason. The protective effect of the oxide layer originates from a low solubility and a slow reaction rate for any chemical interaction between the oxide film and the surrounding environment, the radiolysed water in light water reactors (LWR). The protective effect is, however, never perfect. This means that corrosion of the base material is always occurring to some extent. Furthermore, changes in the chemical composition or pH of the surrounding water, or an increase in temperature, could increase the reaction rate and hence diminish the protective ability of the oxide film.

A slow and well-controlled growth of the passivating film on stainless steels in light water reactors (LWRs) is expected to limit the impact of the coolant on these materials and to minimise the concentration of impurities that may reach the nuclear fuel surfaces and become radioactive. The oxide film forming on stainless steel in reactor conditions has a duplex structure [1–8]. The outer porous layer of the oxide grows via a dissolution/precipitation mechanism, while the inner barrier-like layer grows via a solid-state mechanism [1, 5, 9–11]. The compact inner layer on material surfaces in high-temperature water contains a significant number of ionic defects. Their presence offers routes for ionic species to be transported through the film, making the dissolution of the metal through the film an important process in addition to film growth. The recently proposed Mixed-Conduction Model (MCM) [10, 11] treats the inner, compact layer of the oxide as an unidimensional, finite homogeneous medium which is created and maintained at a certain steady state thickness via the generation, transport and consumption of ionic point defects (interstitial cations, cation and anion vacancies). In that respect, the MCM includes both interfacial kinetics and solid-state transport as rate determining steps for oxide growth and metal cation dissolution through the film. The specific aim of the present paper is to elaborate a procedure for the estimation of the rate constants of the interfacial reactions and the diffusion coefficients of the solid state transport processes in the inner oxide film by using both in-situ electrochemical impedance spectroscopic and ex-situ surface analytical data for oxides formed on stainless steel in a high-temperature electrolyte.

This paper deals mainly with the inner compact oxide layer, which is thought to be the corrosion rate determining part of the oxide. However, the outer porous layer of the oxide is also important in the sense that in LWR conditions they determine the role of adsorption and surface precipitation of e.g. active scpecies onto the oxide surface. A more comprehensive review covering also the role of the outer layer of the oxide film and reactions in it can be found in reference [12].

Experimental

AISI 316 L(NG) stainless steel has been has been studied in a Ti-cladded autoclave at temperatures 150–300°C using a Pt sheet as counter and an external pressure-balanced 0.1 M KCl/AgCl/Ag electrode as reference electrodes. Pure Ir was employed as a metal probe for the measurements in the alloy/oxide/metal configuration. 0.1 M $Na_2B_4O_7$ ($pH_{200°C}$ 8.7, $pH_{300°C}$ 9.1) deaerated with N_2 (99.999%) was used as electrolyte. Impedance spectra in the alloy/oxide/electrolyte configuration were obtained with a Solartron 1287/1260 system in a frequency range 0.02–30,000 Hz at an ac amplitude of 20 mV (rms). Impedance spectra in the alloy/oxide/inert metal configuration were performed using the Ir probe as both counter and reference at zero dc current, ac current amplitude of 10 μA (rms). Auger Electron spectra and depth profiles were measured using a Perkin Elmer PHI model 610 scanning Auger microprobe. The base pressure in the chamber was below 2.67 $\cdot 10^{-7}$ Pa. For depth profiling, sputtering by an Ar^+ ion beam (energy 3 keV, current 2.5 μA) was performed. Sputtering rate was 4.5 nm min^{-1} vs. a Ta_2O_5/Ta sample.

Results

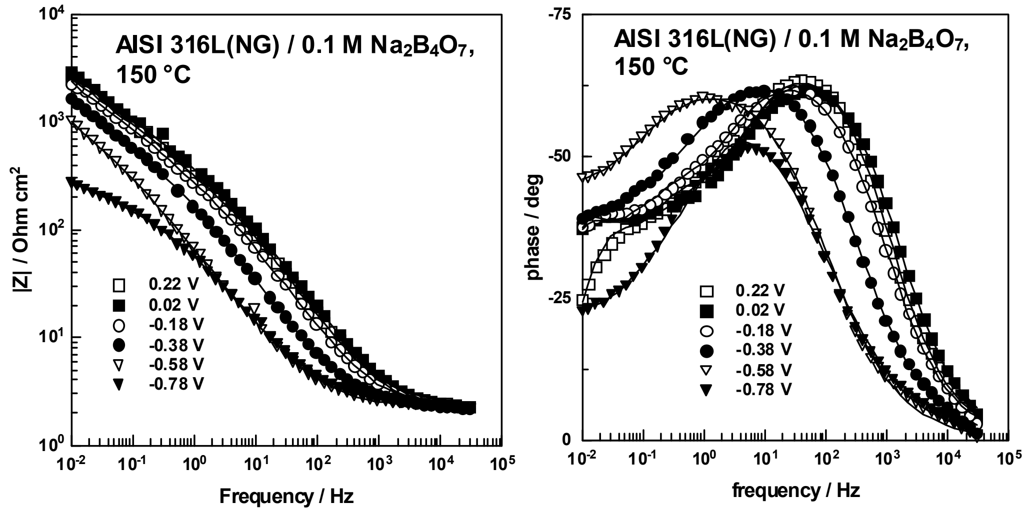

Figure 43. Electrochemical impedance spectra of AISI 316L(NG) in 0.1 M $Na_2B_4O_7$ at 150°C measured in the alloy/oxide/electrolyte configuration. Left – Bode magnitude plot, Right – Bode phase plot. Points – experimental values, solid lines – best-fit calculation according to the procedure outlined in the text.

Figure 43 shows the impedance spectra measured at 150°C after 1 h of oxidation at 0.22 V (at the foot of the transpassive region) and subsequent polarisation with 0.1 V/30 min. steps down to the hydrogen evolution region. Qualitatively similar spectra have been measured at the other temperatures. The impedance magnitude at low frequencies does not depend significantly on potential in the passive region, decreasing slowly both at more negative and more positive potentials. The spectra in the passive region comprise a high-frequency time constant at 1–100 Hz and a low frequency part (0.01–1 Hz) with a phase angle around 45°. To a first approximation, the high-frequency time constant in the spectra can be ascribed to a relaxation in a space charge layer or an interfacial charge transfer process, whereas the low-frequency part has been best described as a Warburg-type transport impedance in a finite-length layer.

Figure 44 shows the impedance spectra measured at 150°C in the alloy/oxide/inert metal configuration after an oxidation procedure similar to that for the spectra presented in Figure 43. The spectra in the alloy/oxide/inert metal configuration comprise a high-frequency time constant (100–5000 Hz) that can also be related to electronic processes in the space-charge layer, and a low-frequency time constant (1–50 Hz) which could be described once more by a finite-length transport impedance. The characteristic frequencies of the transport-type time constant in the spectra measured in the symmetrical configuration are ca. two orders of magnitude higher than those in the spectra measured in the assymetrical configuration (cf. Figure 44 and Figure 45). This means that the transport process associated with this time constant is related to current carriers that attain their steady state transport rate faster than those in the assymetrical configuration. This observation implies that in the alloy/oxide/electrolyte configuration the ionic point defects are the unblocked current carriers, whereas in the alloy/oxide/inert metal configuration this role is played by the electronic defects [11]. Summarising, the picture of the oxide that emerges from the impedance measurements is that of a mixed-conducting film with predominant electronic conductivity. The fact that the impedance of this film does not change significantly with temperature suggests that its electrical and electrochemical properties remain qualitatively unaltered.

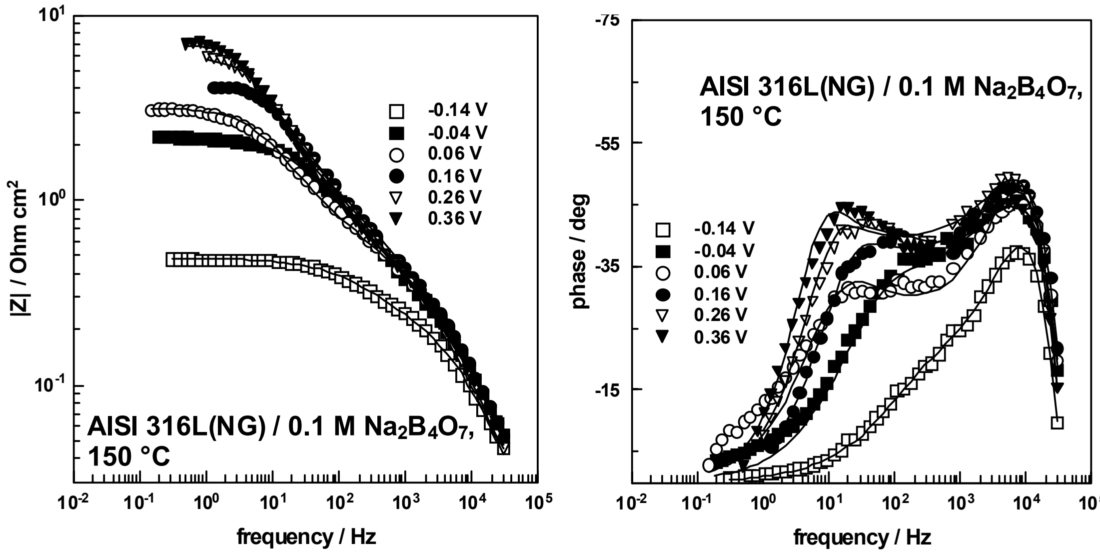

Figure 44. Electrochemical impedance spectra of AISI 316L(NG) in 0.1 M $Na_2B_4O_7$ at 150°C measured in the alloy/oxide/inert metal configuration. Left – Bode magnitude plot, Right – Bode phase plot. Points – experimental values, solid lines – best-fit calculation according to the procedure outlined in the text.

Figure 45 summarises the depth profiles of the atomic concentration of oxygen, as well as the relative concentrations of Fe, Cr and Ni normalised to the total amount of metallic elements for the films formed at 150°C.

An estimate of the film thickness was obtained by computing the depth at which the oxygen concentration drops to half of its maximum value. The estimated position of the alloy / oxide interface is indicated with vertical arrows. The dependence of the oxide film thickness on oxidation potential is non-monotonous, although this parameter in general increases with potential. As an overall conclusion from the surface analysis, it can be stated that the films formed on AISI 316 in 0.1M $Na_2B_4O_7$ at 150°C in the passive region are relatively thin and Cr-enriched, thus having a lot in common with room-temperature passive films. The transpassive film at 150°C is thicker than the passive film, being Cr-depleted and Ni-enriched. More evidence for the duplex structure of the oxide film is found in the surface analysis of the films formed at 300°C, where a Cr enrichment is again observed in the passive region. In the transpassive and secondary passive regions, the films are depleted in Cr and enriched in Fe and especially in Ni. Accordingly, a transformation from a Cr-like to a Ni-like inner layer takes place at the passive-to-transpassive transition.

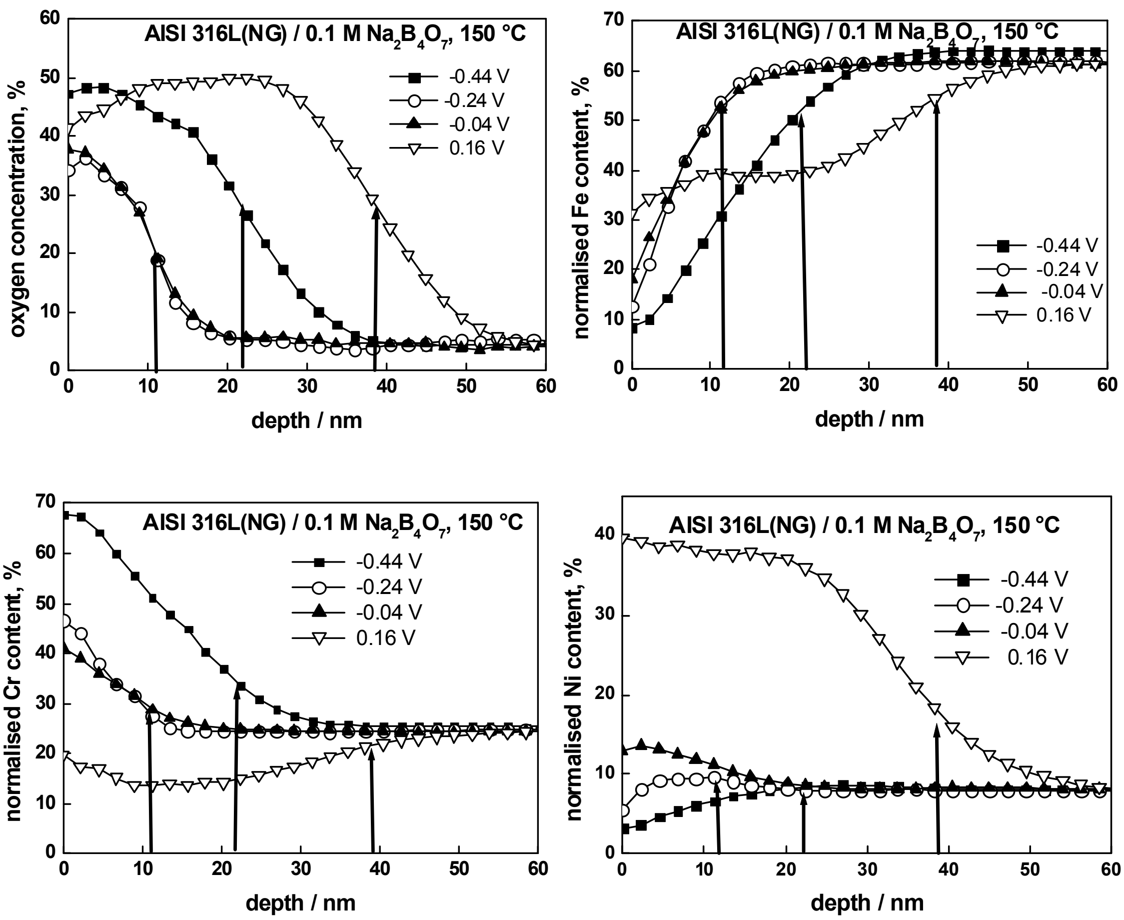

Figure 45. Auger depth profiles of the atomic concentration of oxygen (top left), the normalised contents of iron (top right), chromium (bottom left) and nickel (bottom right) for oxides formed on AISI 316L(NG) in 0.1 M $Na_2B_4O_7$ at 150°C for 72 h. Estimated position of the alloy/oxide interface indicated by arrows.

Discussion

A simplified picture of the typical oxide formed on construction materials in LWRs is presented in Figure 46. To construct this scheme, it is assumed that the oxides that form on stainless steels in these conditions are solely spinels, which is largely consistent with both thermodynamic calculations and experimental observations [1, 2, 8]. In the following, the relationship between the defect structure, ionic and electronic conduction is treated on the basis of the Mixed-Conduction Model (MCM) for oxide films [10, 11].

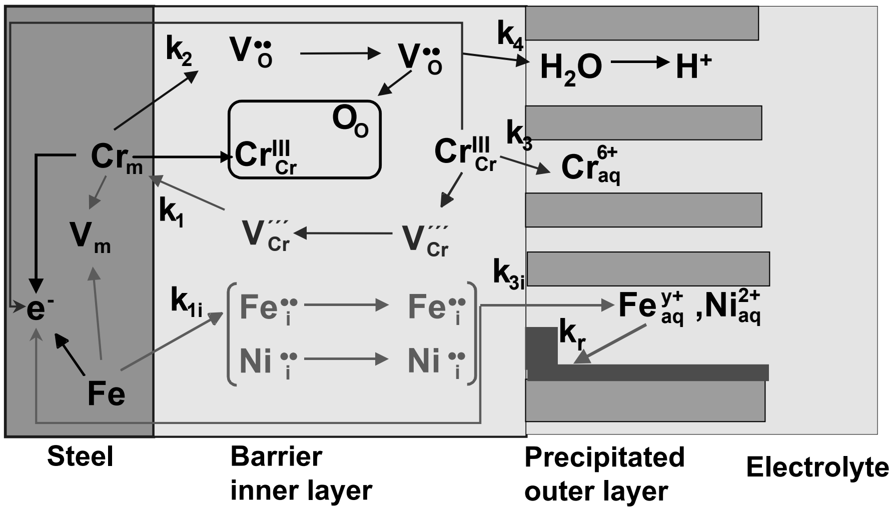

Figure 46. A simplified picture of the main processes taking place during oxidation of stainless steel in high-temperature electrolyte according to the Mixed-Conduction Model.

According to this model, the growth of the barrier layer proceeds into the steel by ingress of oxygen transported via oxygen vacancies (reaction sequence k_2-k_4 coupled via the oxygen vacancy transport flux, Figure 46). The barrier layer is considered to be a normal spinel of the chromite ($FeCr_2O_4$) type [8]. The growth of the barrier layer in high-temperature water is essentially completed in a short time scale. Thus close to steady-state, the sequence k_2-k_4 can be neglected as a slow reaction in parallel to the dissolution of metal through the oxide. The outer layer is considered to be an inverse spinel of the trevorite type ($NiFe_2O_4$). It grows via dissolution-precipitation mechanism, the transport of interstitial cations through the barrier layer being the rate-limiting step (reaction sequence k_{1i}-k_{3i}-k_r in Figure 46 coupled via the transport flux of interstitial cations). The extent of oxide precipitation is governed by the solubilities of Fe and Ni in high-temperature electrolytes, which are rather low, and thus the precipitation reaction is near local equilibrium enabling the formation of large crystallites in agreement with experimental observations [1, 2, 8]. The third process taken into account is the transpassive dissolution of chromium as chromate, mediated by the transport of cation vacancies through the barrier layer (reaction sequence k_1-k_3 in Figure 46 coupled via the cation vacancy transport flux). Summarising, we describe the processes in the oxide in terms of four rate constants, k_{1i}, k_1, k_{3i} and k_3 (the last two dependent on potential with exponential factors b_{3i} and b_3), two diffusion coefficients – D_e and D_i – the field strength in the oxide $\boldsymbol{E}$, the thickness of the barrier layer in which point defect transport occurs, *L,* and the polarisability of the barrier layer / electrolyte interface, α. A global fit of all the impedance data at a given temperature to the transfer function of the model [11] gives the possibility to compute these parameters within an error margin of ±10%. The fitted spectra are shown in Figure 43 – Figure 44 with solid lines and demonstrate the good correspondence between theory and experiment. In addition, using the obtained set of estimates, the quasi-steady state current densities and the thickness of the films on the AISI 316 stainless steel in 0.1 M $Na_2B_4O_7$ at all the studied temperatures have been successfully predicted. The values of the parameters are collected in Table 6.

The values of the standard rate constants and the diffusion coefficients of ionic and electronic current carriers increase with temperature in the range 150–300°C, as expected on general kinetic grounds. The calculation results point to a much bigger influence of the reactions involving interstitial cations on the electrochemical behaviour of stainless steel in the studied temperature range when compared to the reactions involving cation vacancies. The mean electric field strength is ca. an order of magnitude smaller at high temperatures than at room temperature. This can be explained by an increased defectiveness of the film at the higher temperatures. The relatively high values of the diffusion coefficient of ionic defects suggests the predominance of a grain-boundary diffusion mechanism which is in agreement with the presumed nanocrystalline structure of the barrier layer.

Table 6. Kinetic parameters for the oxide growth and restructuring determined by the proposed calculation procedure.

Parameter	150	200	250	300
D_e / cm^2s^{-1}	$1.5•10^{-13}$	$2.0•10^{-13}$	$4.0•10^{-13}$	$6.0•10^{-13}$
D_i / cm^2s^{-1}	$2.5•10^{-15}$	$6.5•10^{-15}$	$0.9•10^{-14}$	$2.5•10^{-14}$
k_{1i} / mol $cm^{-2}s^{-1}$	$3.2•10^{-12}$	$1.0•10^{-11}$	$3.5•10^{-11}$	$6.7•10^{-11}$
k_{3i}^0 / cm s^{-1}	$4.0•10^{-10}$	$6.0•10^{-9}$	$3.0•10^{-8}$	$4•10^{-7}$
α	0.81	0.83	0.85	0.88
$\boldsymbol{E}$ / V cm^{-1}	$2.2•10^5$	$2.0•10^5$	$1.4•10^5$	$0.8•10^5$
b_{3i} / V	7.3	7.7	9	11
k_3^0 / mol $cm^{-2}s^{-1}$	$4.0•10^{-12}$	$6.0•10^{-12}$	$3•10^{-11}$	$7.0•10^{-11}$
k_1 / cm s^{-1}	$7.0•10^{-10}$	$1.0•10^{-9}$	$1.5•10^{-9}$	$4•10^{-9}$
b_3 / V	4.3	4.0	3.9	3.7

Conclusions

The Mixed-Conduction Model treats the inner layer of the oxide film as a unidimensional, finite homogeneous medium which is created and maintained at a certain steady state thickness via the generation, transport and consumption of ionic point defects (interstitial cations, cation vacancies and anion vacancies). The formation of a porous overlayer is modelled as a first-order reaction of reprecipitation. The kinetic parameters of these rate limiting steps are determined in the temperature range 150–300°C by comparison of the kinetic expressions of the model to experimental impedance spectra in two configurations – alloy / oxide / electrolyte and alloy / oxide / inert metal. The obtained estimates can successfully account for both the quasi-steady state current density and the overall film thickness at all temperatures. As a further step in the present investigation, on the basis of the obtained database of kinetic constants and transport parameters, an assessment of the rates of transport and incorporation of minor species in the compact oxide layer is foreseen. In addition, the transport rate of such species through the porous overlayer will be treated as a function of the electrochemically active area at the bottom of pores between crystallites where dissolution / reprecipitation occurs.

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6. Ageing of the function of the containment building (AGCONT)

6.1 Ageing of the function of containment

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Abstract

This research is focused on Nuclear Power Plant (NPP) concrete structures and addresses in general the destructive stresses to which they are subjected. An overall view was also taken into account when selecting constructions and their level of examination during NPP visitations. In addition to examining the NPP containment, some other structures were also included in the study, which were mainly sea structures. This report provides an overview of the main destructive forces on concrete containment structures.

Introduction

Normally the orders of magnitude of the destructive forces to which a concrete structures is subjected to are known and these forces are taken into consideration during the design stage. In any case, the ageing of a structure may depart from the predicted lifetime if the actual destructive forces are not those predicted and e.g. if material properties are not equal to the designed class.

It is well-known that in NPPs, there are massive concrete structures which are also exposed to relatively severe environmental actions.

The most substantial concrete structures in NPPs are heavily reinforcement and many of them are prestressed. Therefore sampling by coring is avoided. Also non-destructive testing is often limited because the structures are so massive and difficult to reach.

Main destructive forces

Chlorides

In NPPs, chlorides are usually derived from seawater. They may also accumulate in concrete during the mixing of concrete with individual components.

The alkalinity within the concrete pore structure provides a protective coating of oxides on the surface of the steel reinforcement bars. However, when chloride ions come into contact with the reinforcing steel they attack the passive layer. Corrosion is initiated at the moment the pore solution chloride content reaches a critical value, so called the threshold value.

In the propagation phase of corrosion, cracks are formed in concrete around the reinforcement. Eventually the corrosion products tend to split the concrete cover and the reinforcement will then lose its function.

The cross-section of the reinforcement bars will get smaller as they corrode. Corrosion at the cracks in concrete may also lead to deterioration due to the reduction of the reinforcement cross-section without splitting of the concrete cover.

Chloride attack is most severe in sea structures at the water-line or in the splash zone. Seawater will enter concrete by capillary action from the water-line upwards. Above the water-line, the water evaporates leaving chlorides concentrated in concrete.

Atmospheric carbon dioxide

Concrete cover provides reinforcement with protection. The chemical corrosion protection of reinforcement will disappear when concrete is carbonated due to atmospheric carbon dioxide. Due to carbonation, the alkalinity (pH) of concrete decreases and the protective coating of oxides may be destroyed. Carbonation of concrete starts from the exterior surface and propagates inward at a relatively uniform front. The propagation is slower with denser concrete.

Carbon dioxide can mainly enter concrete through the air filled pores and it is fastest at about 85–95% relative humidity. Dense coatings on the concrete will considerably retard carbonation as well as dense concrete with a high strength. Carbonation has to be taken into consideration in NPPs, especially in seawater structures above the water level. Carbonation will make reinforcement corrosion considerably faster in concrete containing chlorides.

Sulphates and acids

Sulphate attack involves sulphate ions (SO_4^{2-}) penetrating concrete and reacting with compounds in the cement paste, causing volumetric expansion and thus cracking of the concrete. Cracking will make it easer for additional sulphates to enter the concrete and the structure can entirely lose its strength if there is a continuous influx of sulphates.

In sulphate corrosion, sulphates first react with calcium hydroxide in the hardened cement paste and gypsum accumulates. Then the sulphate ions and calcium react with calcium aluminate hydrates. Deterioration is caused by the formation of ettringite, which has a volume many times greater than the initial compounds. Deterioration requires a high internal concrete humidity for these reactions to proceed.

If water continuously passes through a concrete structure, substantial amounts of sulphates may also accumulate in the concrete. This can cause rapid and complete degradation of concrete if the cement used in the concrete is not sulphate resistant.

In some cases salts, especially sulphates, may also crystallize in concrete. The crystallization pressure may break a concrete structure, usually initiating from the surface. Water alone will also dissolve lime compounds as it passes through concrete over time. The concrete's density and strength properties will be weakened if the water flow through the concrete is not prevented.

Any types of processing acids are harmful for concrete. They are more detrimental at higher concentrations.

In-service temperature

In-service temperatures can play a role in concrete durability. Based on literature information, concrete weakening due to thermal changes will generally not occur at temperatures below +100 °C.

When the concrete temperature is +100 – +200 °C, vaporizable water will escape. This includes the capillary water, water bound physically on the gel pore surfaces and part of the chemically bound water.

At temperatures above +200 °C, more chemically bound water will escape. Hydrated portland cement will dehydrate at +200–500 °C. This will cause intense shrinkage of the cement paste. At the same time the structure of concrete will weaken and it will lose strength. The weakening will be most severe at temperatures between +500 – +600 °C, when the calcium hydroxide is converted to free lime and water. At +850 °C practically all of the cement paste is dehydrated.

There are some structures in NPPs that are constantly at about +300 °C and thus the chemical changes of the concrete must be considered.

The ratio of concrete compressive and tensile strength at elevated temperatures compared to 20 °C is presented in Figure 47 [1].

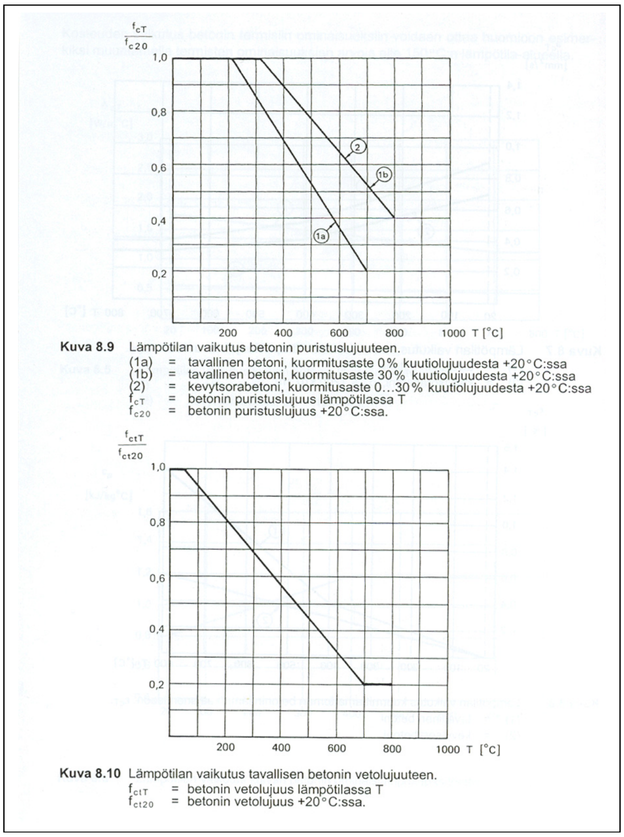

Figure 47. Ratio of concrete compressive and tensile strength at elevated temperatures compared to 20°C [1].

The total volume change of concrete at high temperatures is the sum of the volume change of the cement paste and aggregate. For normal-aggregate concrete, the paste shrinkage is exceeded by the aggregate expansion so there is a net expansion of the concrete.

Rock minerals containing quartz, such as granite, gneiss and quartzite, are all liable to so called quartz-transformation, which takes place gradually between +500 and +650°C and is strongest at 575°C.

This is the reason why concrete containing quartz aggregate is generally considered to be somewhat of inferior-quality at high temperatures compared to other aggregates. For a concrete containing 300 kg/m^3 of cement, the typical thermal expansion of dry concrete containing various aggregates is: 13.0×10^{-6}/°C for quartz, compared to 9.7×10^{-6}/°C for granite and 7.2×10^{-6}/°C for limestone [2]. A lower thermal coefficient of the aggregate is considered beneficial for reducing weakening of concrete properties at elevated temperatures.

Ordinary concrete can be used at temperatures below +250°C, providing that at temperatures constantly above 80°C the design values for the compressive strength and the coefficient of elasticity are experimentally established. For temporary elevated temperatures (< 24 hours), a reduction in the design value can be exploited. In such a case, the reduction factor for compressive strength at +250°C is 0.7 and for the coefficient of elasticity is 0.6 [3].

Based on many investigations, elevated temperatures affect the flexural tensile strength substantially more than the compressive strength. According to [4], at +300°C the flexural tensile strength will decrease to half of the original value (apparent short-term temperature stress). Cyclic temperature stress will be even more detrimental. According to [3], normal concrete should not be used at temperatures above +250°C. Special precautions should be taken or special materials should be used (such as heat-resistant calcium aluminate cement) in such severe environments. The material properties should always be experimentally verified to determine the concrete's thermal properties before use.

Ageing of prestressing steel tendons

With age, the effectiveness of prestessing steel tendons changes due to relaxation. According to [1], the value for relaxation is, with certain presumptions, three times greater than the value measured in a 1000 hour relaxation test. A temperature rise will increase the relaxation. For instance, according to [5] the relaxation will increase about 50% when the temperature increases from +20°C to +25 – +40°C.

A rise in temperature above +100°C will also reduce the ultimate strength of the prestressing steel tendons. Until +200°C, the reduction is below 10%, after which time the loss needs to be verified.

Creep of concrete

Concrete creep will reduce the tension in prestressing steel tendons. Creep depends on many factors and it is not possible to estimate it exactly. It depends e.g. on concrete composition, strength, degree of hydration and humidity at the time of prestressing and afterwards during the use. The following aspects will reduce creep:

- concrete with a low water-binder ratio and high strength
- using hard aggregate with a high modulus of elasticity value
- using a coarse gradation of aggregate
- later age and higher strength at first loading.

Concrete drying over the course of loading will considerably increase creep. For instance, in the Finnish concrete standards the creep-coefficient for moist concrete is 1, but for dry concrete it is 3 (40% RH). At a relative humidity of 90% RH the creep-coefficient is 1.5. [1]

High temperatures will also increase creep, e.g. at +70°C creep is about 3.5-times the value at +20°C [6]. Concrete creep will not increase considerably after 10–20 days of loading if the concrete is not allowed to further dry.

Drying shrinkage of concrete

Concrete drying shrinkage is affected by many factors, the most important being the water content of the mixture. Concrete mixtures with a higher water content, and thus usually a higher paste content and higher water-to-cement ratio, will have greater drying shrinkage. Using larger amounts of coarse aggregate, and aggregate with a larger maximum aggregate size, will reduce drying shrinkage.

Concrete shrinks less in a more humid environment. A massive concrete structure will shrink less than a thin structure, but the shrinkage will continue for a longer period. The amount of shrinkage is dependent on restrain provided, such as by the substrate or adjacent walls and columns. Drying shrinkage is less in reinforced concrete than plain concrete, depending on the amount of reinforcement. In prestressed structures, drying will reduce tension in prestressing tendons.

Shrinkage continues for many years. Drying shrinkage will lead to cracks if not properly accounted for during mixture and structural design, i.e. by providing joints in a slab. Cracking affects concrete durability, reinforcement effectiveness and fastness of coatings.

Concrete volume changes will also cycle with normal temperature and moisture fluctuations, for instance a hardened concrete will slightly expand with a gain in moisture.

Coatings

During an accident, concrete coatings inside a NPP containment will be stressed in ways which highly differ from the stresses during normal operation. Under accident conditions both coatings on walls and floors will be affected. Coatings on floors are more critical as they are normally thicker and thus embody more material.

The detailed requirements for coatings inside containments are given in [7]. The requirements are given for the following issues specified in Guides YVL 4.1 and YVL 4.2:

- radiation resistance
- ease of decontamination (decontaminability)
- chemical resistance
- durability under operating conditions
- durability under postulated accident conditions
- fire technical properties.

The requirements are based on the referred standards and on the Nordic TBY-criteria.

Conclusions

Concrete structures in NPPs are subjected to highly variable circumstances. These environmental conditions are however fairly well known as well as the response of concrete structures to these conditions. It can be assumed that unexpected ageing processes or risks do not exist if properly accounted for during design. However, some NPP concrete structures are exposed to such severe stresses that they must be prepared or sheltered to ensure their service life.

It is very difficult and actually impossible to continually examine all NPP structures to provide detailed recommendations for their treatment. It is only possible to conduct some tests and assessments at regular intervals to ensure that containment constructions will meet all demands in any possible accident. For instance, testing of prestressing tendons can be done using pressure and leakage tests to experimentally verify in-situ performance.

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7. Concrete technological studies related to the construction, inspection and reparation of the nuclear power plant structures (CONTECH)

7.1 CONTECH summary report

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Abstract

The subtasks of CONTECH deal with concrete from mix design to the end of service life of a structure. So far both design directions and directions for site to avoid cracking of concrete structures have been prepared, a tool for optimising the management of concrete structures has been created, chloride intrusion into concrete has been clarified, quality requirements for milling of concrete has been developed and the maximum allowable extent of rust on reinforcing bars for new structures has been determined.

Introduction

CONTECH consists of several tasks the results of which are applicable on the design of prestressed and nonprestressed concrete structures, development of inspection and reparation methods, controlling of ageing behaviour and getting prepared for and controlling of accidents.

Main objectives

Main objectives have been preparing directions for the prevention of cracking, creating tools for economic management of structures, clarifying the theory of chloride ingress into concrete, defining optimal milling technique and determiming the allowable extent of rusting of reinforcing bars for new structures.

To determine extra costs caused by the delay of repair optimum timing, the life cycle costs and optimum timing for repair of different structures were determined. Also the extra costs caused by delays to this timing were determined. The comparison of maintenance strategies was based on life cycle cost analysis in which the following calculation assumptions and principles were applied:

- costs based on realized reparation and maintenance contracts
- existing degradation models and service life information
- generally accepted calculation methods of life cycle costs
- life cycle cost analysis carried out based on Markov chain.

Figure 48 shows the principle of life cycle cost analysis.

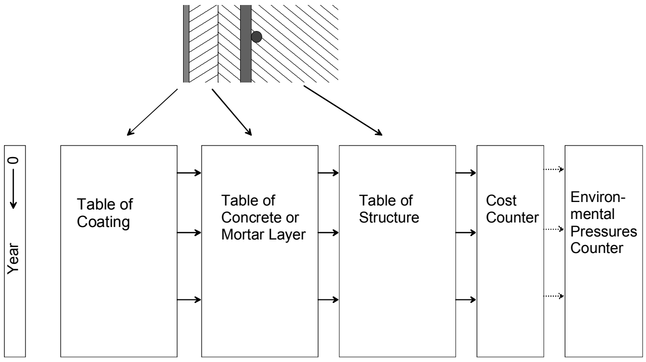

Figure 48. The principle of life cycle cost analysis.

A literature survey was carried out to study the penetration and leaching of chlorides into and out of concrete. This is essential to be known to determine the optimum service life and timing of reparation.

Field experiments were carried out to determine the quality requirements of a milled surface concerning roughness, cracking and tensile strength of the surface.

Reinforcing bars, A500HW, were rusted in laboratory and different kinds of rust removal techniques were tested on them.

Main results

When minimizing cracking of concrete structures it is essential to recognize the risks of cracking both in design and construction. Critical places for plastic settlement cracking are upper parts of columns and wall like structures, thick structures and joints of structures with different thicknesses. There is an increased risk of plastic shrinkage cracking when concrete strength is higher than 40 MPa, concrete has a long setting time and water evaporation rate from the concrete surface is high. Cantilever like structures tend to crack when L/H ≥ 2 … 2,5, high strength concrete is used or the temperature is high or low during casting.

Factors that affect cracking risk during concreting are size of structures cast at one time, placing of construction joints, temperature fields and strength development, concreting schedule, weather conditions during concreting, curing and strength of concrete during demoulding. Factors that should be taken into account during construction are timing of concrete deliveries, distance between casting layers, right placing of temperature measurements, timing of vibration and re-vibration, using of evaporation retarder and timing of screeding, curing and demoulding.

The three main factors affecting the chloride penetration are the material properties of concrete, manufacturing technique of concrete and environment.

Figure 49 shows the principle of the humidity and chloride content profiles in the convection and diffusion zones.

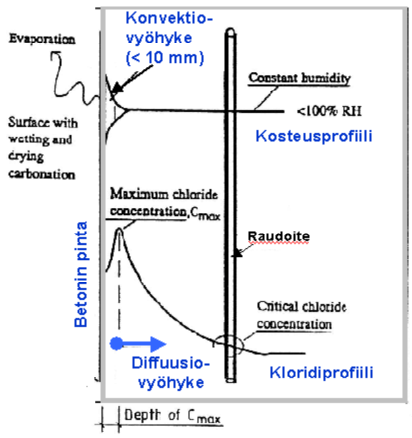

Figure 49. The principle of the humidity and chloride content profiles in the convection and diffusion layers (Sandberg 1998) [1].

Figure 50 shows the correlation between milling depth and surface tensile strength.

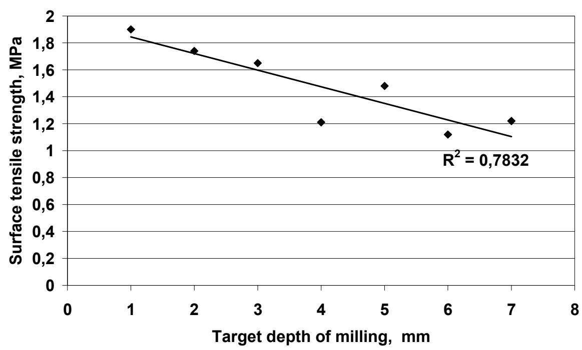

Figure 50. Correlation between surface tensile strength and milling depth.

As the milling depth increased there was also an increase in both macroscopic and microscopic roughness, areal portion of the cracked surface and depth of cracking.

Four corrosion classes for new structures were developed. Photo illustrations showing a rebar before and after wire brushing are used to determine the corrosion class of a bar.

The corrosion classes are:

0 Acceptable. Almost no rust at all. No need to be wire brushed for inspection. No need to be cleaned before placing.

1 Acceptable. Light adherent rust and loose powdery rust. No need to be wire brushed for inspection. No need to be cleaned before placing.

2 Acceptable. Loose flaky rust. Some local corrosion spots the depth of which is ≤ 2,5% of the nominal diameter of the bar. Before palcing the rust must be removed by shot blasting or pressure cleaning. After placing the reinforcement may be cleaned only as specially agreed with the orderer.

3 Unacceptable. Plenty of loose rust. The depth of local corrosion spots > 2,5% of the nominal diameter of the bar.

Figure 51 shows a wire brushed rebar of corrosion class 3.

Figure 51. Wire brushed rebar, ø 12 mm, of corrosion class 3.

Applications

The results are directly applicable for the design, construction and maintenance of concrete structures.

Conclusions

Main conclusions from the subtasks are:

- Directions of how to minimize cracking risk all throughout the process from design to construction were prepared. In the structural design it must be ensured that the structure behaves as calculated. When a cracking risk exists, measures to prevent cracking must be presented in the concreting plan. The supervisor must ensure that the measures planned are fulfilled.

- To minimize life cycle costs the repairs should be done timely and systematically. It is profitable to repair some structures on the safe side because of the high reparation costs.

- Excluding the outermost convection layer of the structure the penetration of chlorides can be simplified as diffusion. The chloride profile depends on both environment and quality of concrete.

- The target depth of milling is most favourable to be set as low as possible.

- Four corrosion classes were developed. Illustrations showing a rebar before and after wire brushing are used to determine the corrosion class of a bar.

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1. Lindvall, A. 2002 A. Mapping of the chloride load around two Swedish reinforced concrete bridges. Publication P-02:2, Arb. nr 612. Chalmers University of Technology. Department of Building Materials. 93 p. Avalaible:http://www.bm.chalmers.se/research/Acrobat/p022.pdf

8. Wall response to soft impact (WARSI)

8.1 WARSI summary report

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Abstract

Preliminary calculations for planning the tests in the IMPACT project were carried out with ABAQUS code. LSDYNA is used for analyses involving more accurate fluid-structure interaction and capable of simulating the bursting of a tank. Finally, numerical results will be verified against experimental data.

Introduction

The events of September 11th have emphasized the need to design protective structures for important constructions against external impact loads, caused e.g. by colliding vehicles, and to analyse carefully the possible consequences of such events, taking into account the existing experience and information.

In the open literature there are some fairly well documented test results on the subject of deformable missiles to be used as references in developing and calibrating the finite element simulation models and assessing the obtained numerical results. However, test results for fluid filled soft projectiles are not available in the open literature. In order to get any confidence with the simulation results, experimental, recorded data is needed for verification. The tests will be carried out in the IMPACT project.

Main objectives

The main aim of the project is to develop and take in use methods for predicting response of reinforced concrete structures to impacts of deformable projectiles that may contain combustible liquid, such as jet fuel. Loading, structural behaviour, like collapsing mechanism and the damage grade, will be predicted by simple hand calculations and using non-linear FE-method. For numerical analyses ABAQUS and LSDYNA codes will be used. The spread, penetration and dispersion of fuel in the impact will be predicted.

Main results

1. Pre- and postcalculations for tests

This subproject aims at evaluating the behavior and collapse of different kinds of deformable projectiles, such as fluid filled soft missile, simulating fuel tank. The possible fluid hammer effect will be studied.

A high velocity impact with an aircraft against a rigid building causes a massive transient force to the building. The resultant force depends in some degree on the stiffness of the aircraft. In a MSc Thesis at TUT Erik Eriksson studied the axial crushing force properties of a mid and a large size aircraft. Based on these studies a simulation model to be used as a deformable missile in the IMPACT tests was designed. The essential properties of the missile, like mass and stiffness distribution, correspond to a commercial aircraft. These properties are defined using Finite FE analyses and available data of a commercial aircraft.

In order to test the missile testing equipment some preliminary tests were carried out with a simple inexpensive missile against a rigid steel plate target. Precalculations and test results are presented more detailed in the accompanying Special Report.

2. Droplet formation of liquid fuel

In aircraft crash to a solid surface (such as building wall), certain part of the fuel release from the fuel tanks is dispersed forming an aerosol (droplet) cloud in surroundings. Fuel jet impingement on any solid surface may further lead to the secondary breakup of liquid jet enhancing the atomization. The droplet size and size distribution formed affect the fuel dispersion and the burning processes inside the cloud.

The major fundamental processes of fuel dispersion (mist cloud/pool) and burning must be known in order to assess the external fuel hazards in the plant area in aircraft crash situations. No quantitative, empirical data of large-scale is available from the fuel atomization under real aircraft crash conditions. A literature review is made of the main processes affecting the liquid atomization, and related droplet size and size distribution, based on the available theoretical end empirical experiences e.g. from spray/sprinkler – technology. Applying the empirical and semi-empirical formulas available from literature, rough estimates of atomization of kerosene liquid will be made.

3. Structural response

Firstly, capabilities of LSDYNA code in simulating nonlinear behaviour of impact loaded reinforced concrete structures are tested by comparing the numerical results with the available experimental measurements. Numerical model and von Mises stress distribution in the missile when the missile penetrates the target are presented in Figure 52.

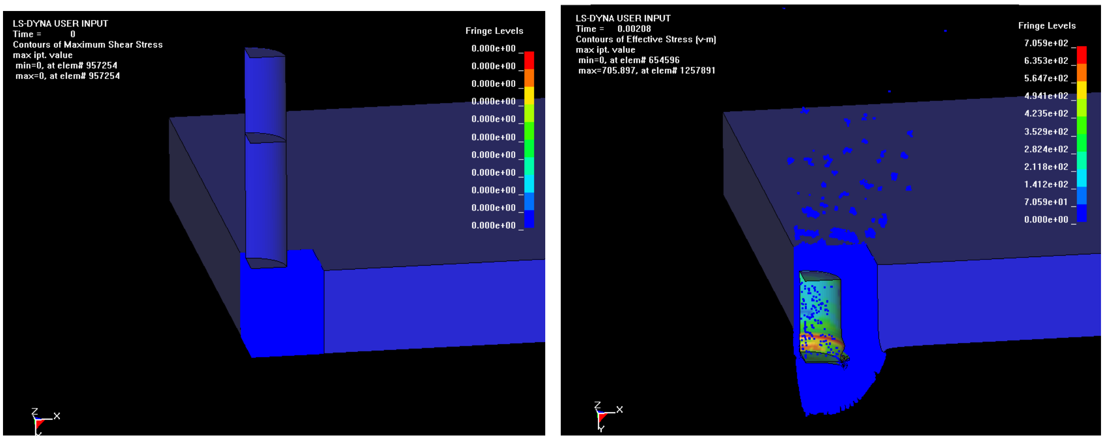

Figure 52. LSDYNA model.

Preliminary calculatios for designing the reinfoced concrete targets were carried out using simplified methods. Also materially nonlinear FE -analyses were carried out in order to predict the optimal locations for the strain gages in the first reinforced concrete targets. These targets will be prepared and tested in 2005.

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8.2 Impact loaded structures

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Abstract

An experimental apparatus has been constructed to simulate the impact of an aircraft against a nuclear power plant. In order to test the equipment some preliminary tests were carried out with simple inexpensive missiles against a rigid steel plate target. Precalculations for the tests were carried out using initial velocities of 100 m/s, 150 m/s and 200 m/s for the test missile.

Introduction

The events of September 11th have emphasized the need to design protective structures for important constructions against external impact loads, caused e.g. by colliding vehicles, and to analyse carefully the possible consequences of such events, taking into account the existing experience and information.

In the open literature there are some fairly well documented test results on the subject of deformable missiles to be used as references in developing and calibrating the finite element simulation models and assessing the obtained numerical results. However, test results for fluid filled soft projectiles are not available in the open literature. In order to get any confidence with the simulation results, experimental, recorded data is needed for verification. The tests will be carried out in the IMPACT project. Numerical analyses needed for planning the tests and measurements are carried out in the WARSI project. The main goal of the WARSI project is to develope and calibrate finite element simulation models for reinforced concrete structures impacted by deformable missiles filled with fluid.

Numerical studies on a semihard missile for preliminary tests

In order to test the missile testing equipment some preliminary tests were planned to carry out with a simple inexpensive missile against a rigid steel plate target.

The main aim is to test the capacity of the test equipment, especially whether the intended velocity for a planned deformable missile is reachable. The mass of the deformable missile is planned to be about 30 kg. A simple pipe profile was chosen for the preliminary semihard missile. The dimensions of the pipe cross-section are presented in Table 7.

Table 7. Properties of the pipe cross section.

Diameter (outer) [m]	Wall thickness [m]	Mass [kg/m]	Area [cm^2]
0.273	0.005	33.1	42.1

The length of the pipe missile is 0.9 m and there is a welded circular cover at the impacting end of the pipe. The thickness of the cover is assumed to be also 5 mm. The weight of the missile used in the preliminary numerical simulations would be 32.3 kg.

Initial velocities of 100 m/s, 150 m/s and 200 m/s were used in analysing the impact of the missile against a rigid wall. Both material and geometrical non-linearity were taken into consideration in finite element (FE) analyses. Nonlinear material properties are presented in Table 8. The Young modulus was assumed to be 206000 MPa and Poisson's ratio 0.3, respectively.

Table 8. Stress vs. plastic strain.

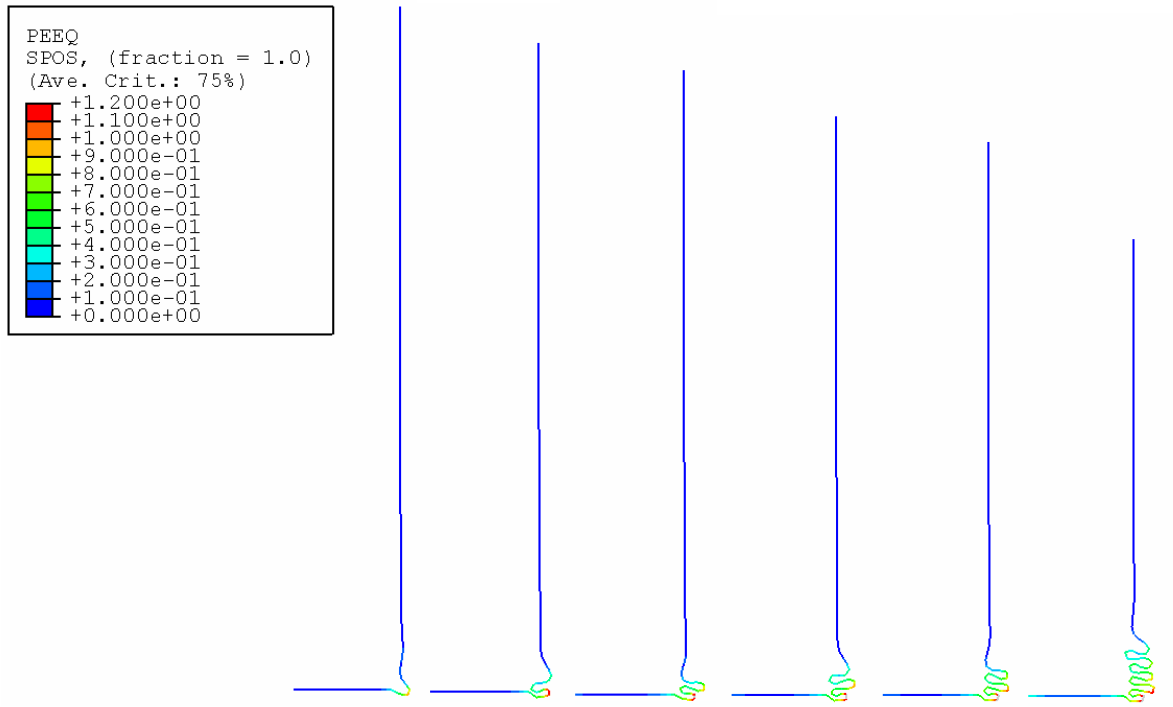
Stress [MPa]	220.	235.	340
Plastic strain (mm/mm)	0.	0.002	0.24

The effect of strain rate to affect the mechanism of plastic flow was also considered. A standard procedure for considering strain rate effects is given by the formula

$$\dot{\varepsilon}_{pl} = D[\frac{\tilde{\sigma}}{\sigma^0} - 1]^p,$$

where $\dot{\varepsilon}_{pl}$ is the equivalent plastic strain rate, $\tilde{\sigma}$ is the effective yield stress and σ^0 is the static yield stress. For structural steels, the material parameters D and p typically assume the values 40 and 5, respectively.

Finite element analyses were carried out using ABAQUS/Explicit code /1/. There are about 100 type SAX1 axi-symmetric elements modelling the pipe wall. The length of one element is slightly less than 10 mm. Deformed shapes during the impact and the amount of plastic deformation at the outer surface of the missile are shown in Figure 53, in the case of an impact velocity of 200 m/s. The calculated reaction force against the target is presented in Figure 54 and the axial displacements as a function of time are shown in Figure 55. The maximum displacement occurs at the end of the missile being about 32 cm. The corresponding maximum displacements with the initial velocities of 150 m/s and 100 m/s are 21 cm and 10 cm, respectively.



(a) (b) (c) (d) (e) (f)

Figure 53. Plastic deformations with the initial velocity of 200 m/s, (a) t = 0.025 ms, (b) t = 0.050 ms, (c) t = 0.075 ms, (d) t = 0.100 ms, (e) t = 0.125 ms and (f) t = 0.300 ms.

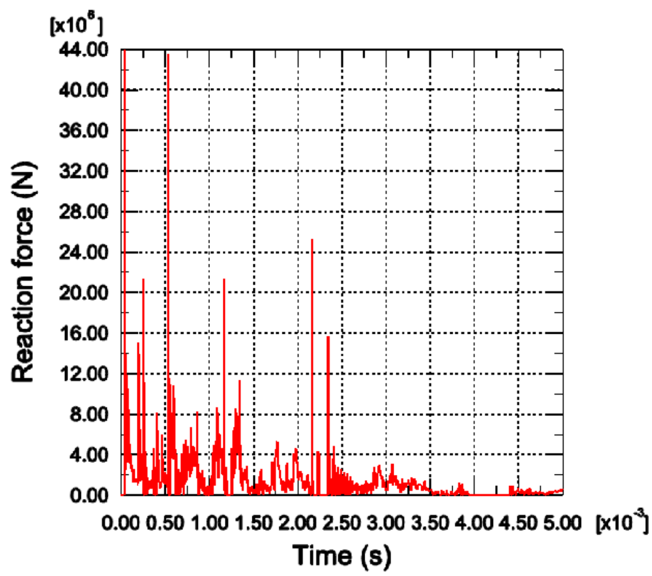


Figure 54. Calculated reaction force as a function of time, v_o=200 m/s.

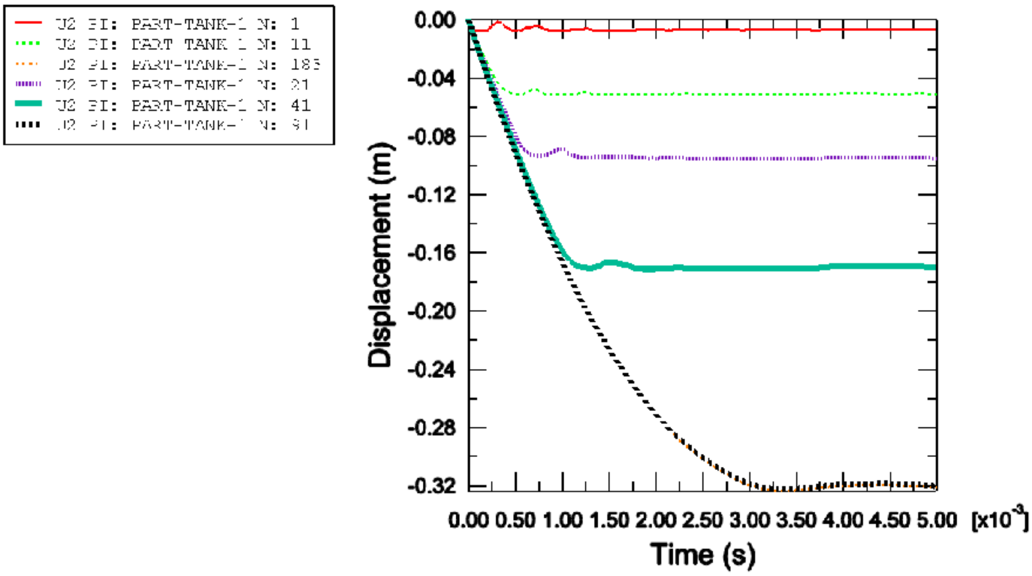


Figure 55. Axial displacements as a function of time, initial velocity 200 m/s.

Impact-force time histories calculated by the initial velocities of 100 m/s and 150 m/s are shown in Figures 56 and 57.

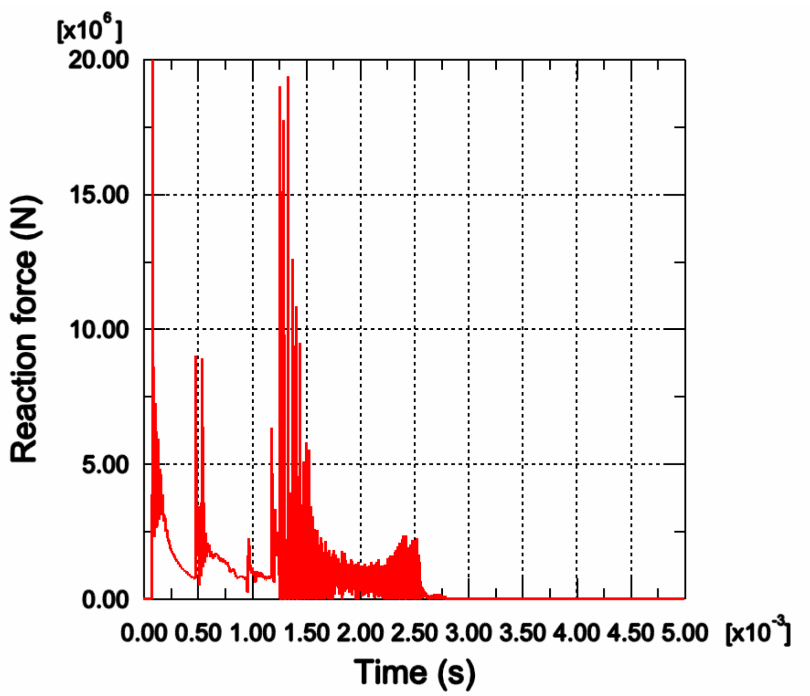


Figure 56. Calculated reaction force as a function of time during the impact, v_O=100 m/s.

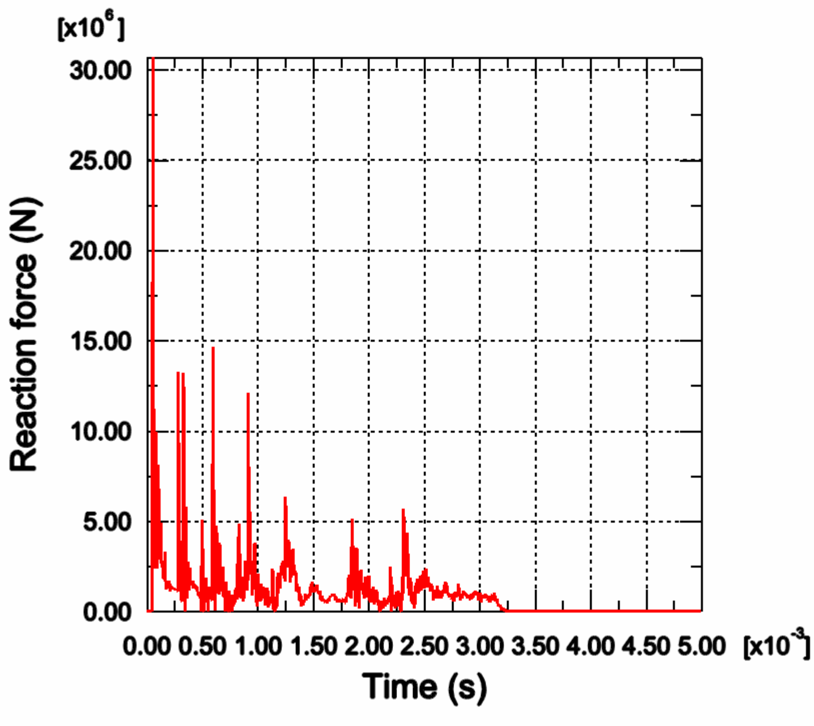


Figure 57. Calculated reaction force as a function of time during the impact, v_o=150 m/s.

A method for calculation the time dependent force resultant F(t) imposed by a deforming missile on a rigid target was reported by Haley and Turnbow in /2/.

Assuming that the mass of the crushed part of the missile moves with the target structure an equation for the conservation of momentum for the missile/target system yields

$$F(t) = \frac{d}{dt}(M_r v_m + M_c v_t),$$

where F is the reaction force, M_r is the mass of the uncrushed part, M_c is the mass of the crushed part, v_m is the velocity of the missile at time t and v_t is the velocity of the target at time t. If the velocity of the target v_t can be neglegted then the above formula yields

$$F(t) = P_c(x(t)) - m(x(t))(v_m(t))^2,$$

where P_c is the crushing load or buckling load of the missile body, m(x(t)) is the mass per unit length of missile (at time t in contact with the target) v(x(t)) is the velocity if the undeformed (or uncrushed) part of the missile at time t.

According to the impact force formula also the crushing force contributes to the resultant force. Crushing forces calculated according to a method presented in /3/ for impact velocities 100 m/s and 200 m/s are indicated in Figure 58 by (P_{cj}, v_o=100 m/s) and (P_{cj}, v_o=200 m/s). The corresponding reaction forces are indicated by (F_j(t), v_0=100 m/s) and (F_j(t), v_0=200 m/s). The corresponding reaction force due to an impact with the initial velocity of 150 m/s is indicated with (F_j(t), v_0=150 m/s).

The curve indicated by F(t) presents the reaction force for an impact velocity of 200 m/s impact velocity calculated by using the plastic yield force as a crushing force. The plastic yield force P_c is obtained simply by multiplying the pipe cross section area by the yield stress.

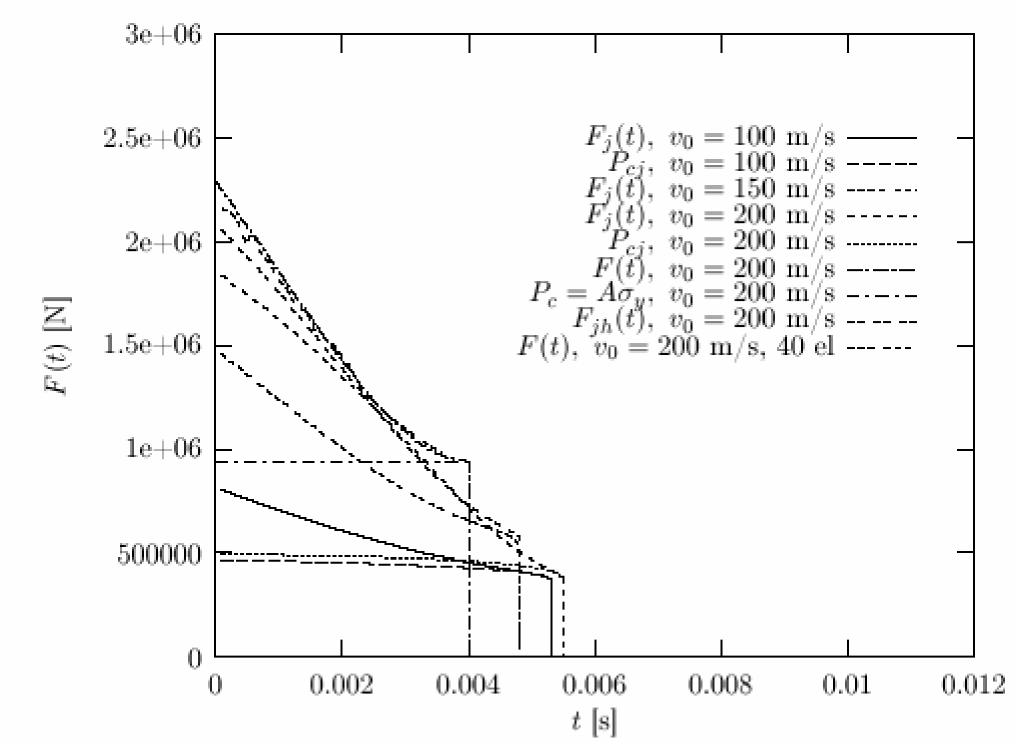


Figure 58. Reaction forces as a function of time according to simplified methods.

Yield hardening of the material affects the crushing force. Reaction force due the impact with a velocity of 200 m/s and using the yield hardening material properties is indicated with F_{jh}.

Impact force time history can also be determined by a model in which both the missile and the target are first discretized as in a one-dimensional finite element or mass-spring model. When a strain in a spring element becomes ε = -1, the masses joined by that spring are assumed to impact plastically. In modelling the impact with a velocity of 200 m/s the length of the pipe was divided to 40 parts. The calculated impact-force time history is indicated in Figure 58 by (F(t), v_0=200 m/s, 40el) /4/.

Experimental simulations

An experimental apparatus has been constructed to simulate the impact of an aircraft against a nuclear power plant. Figure 59 shows the main components of the test set-up. The system was built in research space where VTT has a powerful pneumatic compressor. This devise provides compressed air which is stored in the pressure accumulator shown in Figure 59.

Another important part of the apparatus is the acceleration tube. A strong membrane is installed between the pressure accumulator and the acceleration tube before a test. The membrane is calibrated to brake at a certain level of pressure. Hence, the tests are made by allowing the pressure to increase inside the accumulator until the membran fails. Compressed air starts then flow at a high speed to the acceleration tube where it meets a piston. The compressed air dives the piston towards the impact wall at an increasing speed.

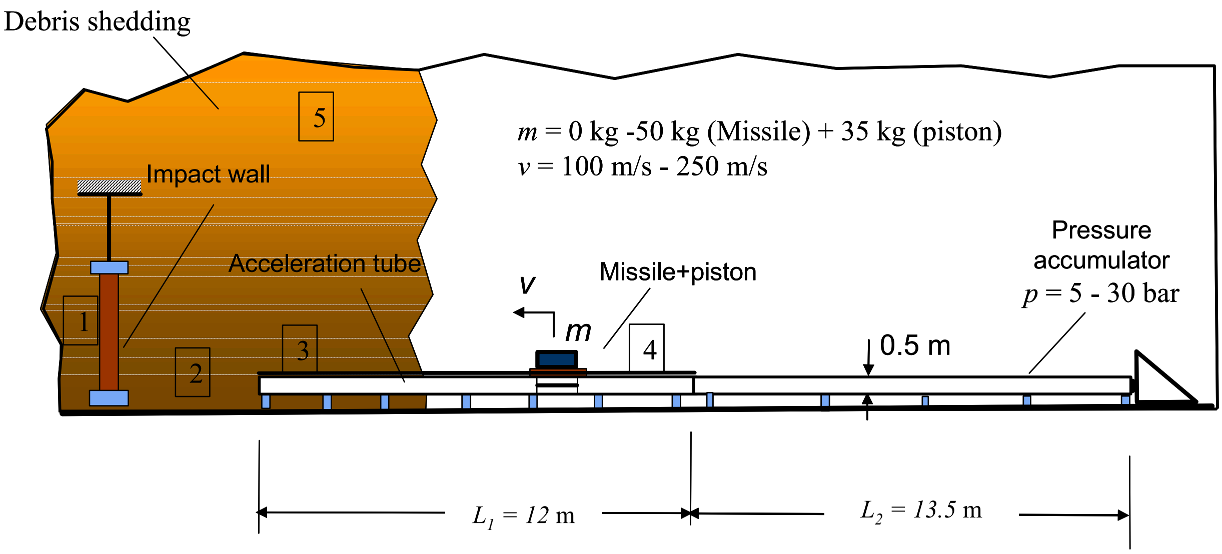


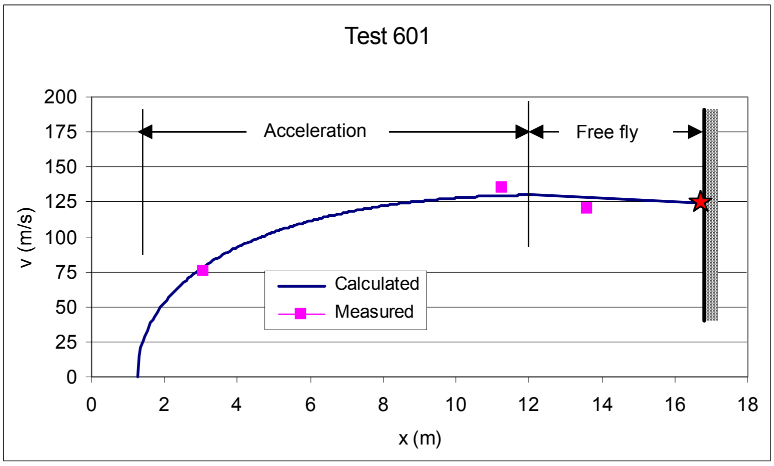
Figure 59. Test set-up for impact tests. Measurements: Acceleration and stresses (1), Velocity of the missile (2), Pressure (3, 4), high speed video (5).

The piston is connected to a missile that stands on a rail above the acceleration tube. The connection is made through a narrow slot. The missile follows the motion of the piston to the end of the acceleration tube. At this position, the piston is forced to fly towards the floor where it crushed inside an inclined plate that is not shown in Figure 59. The missile continues its flight against the impact wall. Figure 60 shows the 25.5 m long apparatus inside an underground space.

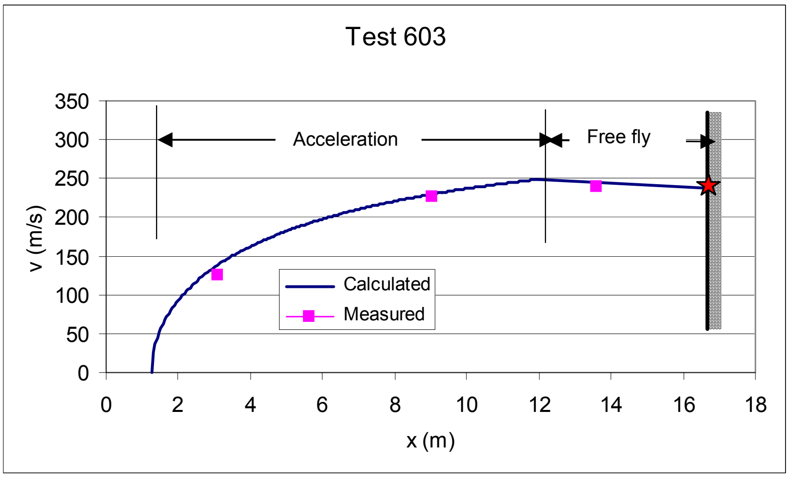
Figure 60. Apparatus for impact tests, installed in an underground space.

The design calculations of this apparatus were made by small computer program that is based on numerical time integration. Equations of gas dynamics were not explicitly used. Instead, the Boyle-Mariotte law was used to roughly estimate the force acting on the piston while the space occupied by the gas increased when the piston moves forward. The diameter of the piston is 5 mm smaller than the diameter of the acceleration tube. Therefore, some of the compressed air leaks. An even larger source of leakage is the 5 mm wide slot above the acceleration tube. The effects of these leakages were considered using a theory developed in /5/.

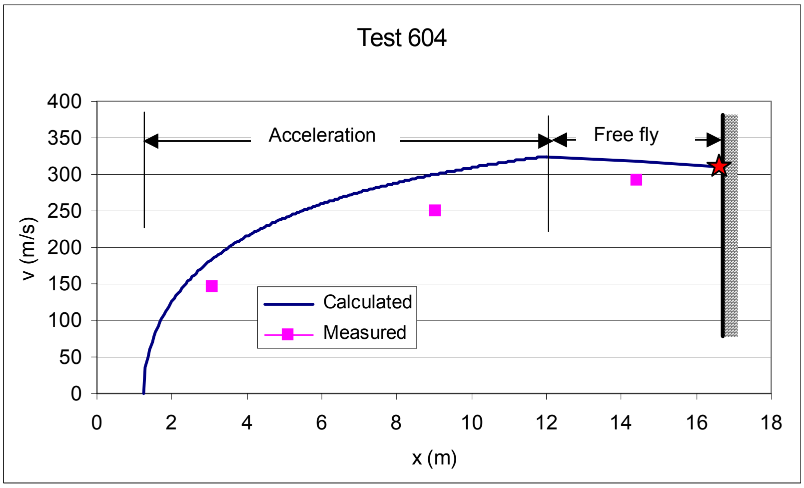
Due to the approximative nature of the design calculations, the apparatus was tested first by making tests with the sole piston. No missile was used in the three first tests. Figure 61 shows comparisons between the measured and predicted velocities. A comparison shows that the agreement is very good at conditions where the impact velocity is lower than 200 m/s. At higher velocities the simplified model predicts too high velocities for the missile.



(a) Initial pressure 3.3 bar



(b) initial pressure 9.6 bar



(c) Initial pressure 17.5 bar

Figure 61. Comparisons between measured and predicted velocities. The weigh of the piston was 23 kg. No missile.

Figure 62. Deformed preliminary missile.

In the first test with a missile the impact velocity was slightly over 100 m/s. The deformed missile is shown in Figure 62. The axial deformation at the end of missile is about 10 cm. This deformed shape corresponds well the preliminary numerical results.

Conclusions

The first test with a simple missile against a rigid steel plate target was conducted successfully. Materially and geometric nonlinear axisymmetric finite element analyses were carried out in order to study the behaviour of a semihard missile due to an impact against a rigid target. Numerical results concerning the first missile test were in agreement with the experimental findings. Analyses of the measured data is still going on.

References

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9. The integration of thermal hydraulics (CFD) and finite element (FEM) computer codes in liquid and solid mechanics (MULTIPHYSICS)

9.1 The integration of thermal-hydraulics (CFD) and structural analysis (FEM) computer codes in liquid and solid mechanics (MULTIPHYSICS)

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Abstract

The main objective of the project is to improve modeling capabilities in fluid-structure-interaction (FSI) systems. Because FSI -problems are very common in industry, there is a need to develop simulation tools for these problems. An important application in nuclear industry is the structural integrity of nuclear reactor core internals in Design Basis Accident (DBA). The specific case studied in the project is Large Break Loss of Coolant Accident (LBLOCA) and especially the pressure transient, which is caused by the guillotine pipe break in LBLOCA. Tools planned to be used for the analysis are computational fluid dynamics (CFD) and finite element stress and strain analysis (FEA) codes. An original aim of the project was to study the possibility to develop a common numerical model for system geometry with suitable meshing for all physical analysis disciplines, including both CFD and FEA modeling. However, quite soon the target proved to be unrealistic. The common meshing would decrease the quality of both analyses too much. In the next phase the target was set on separate calculation of the pressure transient with CFD and FEA codes and also on development of the method, which could link CFD and FEA calculation results through interpolation. Firstly, one way coupling between CFD and FEA has been tested, but the final goal is to use a two way link. The codes used in this work are NASTRAN and ABAQUS on the FEA side and STAR-CD in the CFD side. In this paper separate pressure transient calculations using CFD and FEA codes in LBLOCA case are presented. The case has been calculated separately with NASTRAN and STAR-CD. Also one way link between STAR-CD and ABAQUS is tested.

Introduction

The LBLOCA is an important DBA case in nuclear industry. In Finnish nuclear safety requirements it is stated that the reactor internals must be able to keep their structural integrity in a case of the LBLOCA. The accident case is a FSI problem and it is complicated to analyse it by computational methods. Linking of CFD and FEA codes is a promising method of analysis in a case of FSI problems. However, the problem is that both codes have relatively strict method sensitive demands for qualities of calculation meshes. A common mesh between CFD and FEA would allow savings in the mesh generating process and would also make data transfer between these two methods easier. Unfortunately the quality demand means that similar meshes can not be used, if the accuracy of calculations has to be kept on a high level. The only possibility is to link CFD and FEA codes through interpolation.

A FSI phenomenon means that forces caused by the fluid flow are moving the structures. The significant quantity is the pressure field of the fluid flow, which can be calculated using CFD. The pressure field calculated by CFD can be transferred to the FEA code where structural calculations can be made. This is called a one way link and if the effect of the pressure field on the structure is static, this is sufficient. However, most cases are not static. Often the pressure field is moving the structures, which has an effect on the flow and pressure fields. So, a two way link is needed. The two way link is quite complicated in practice especially when the mesh structures in CFD and FEA models are different. The problem is not just communication between CFD and FEA codes, but the results must be also be interpolated to a different kind of mesh structure. The first step in this project has been to make pressure transient calculations separately with CFD and FEM codes. The first phase has improved the understanding of modelling capabilities of both CFD and FEM codes. In the second phase linking of the codes has been started performing a one way link calculation.

NASTRAN analysis

Msc. NASTRAN is a commercial FEA code, which is used in the project to calculate the pressure transient inside the reactor pressure vessel in the LBLOCA case. Calculations are performed by ENPRIMA. The NASTRAN model used in the project is based on the model, which includes whole primary loop of Loviisa Nuclear Power Plant (NPP). The model is presented in Figure 63. However, in this work only the reactor pressure vessel with its internals was included in the calculation. The pressure drop was given as a boundary condition on the pipe cross-section surface just before the cold leg pipe inlet to the reactor. The location is the supposed rupture point of the reactor inlet pipe in LBLOCA.

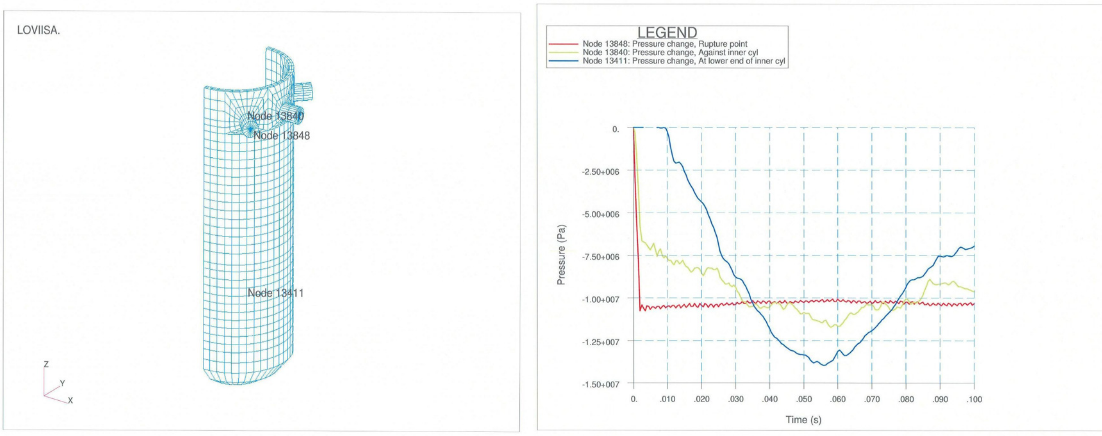

Figure 63. NASTRAN model used in the project and pressure profiles at three points inside the reactor calculated by NASTRAN.

The model comprised both fluid and structural parts. Pressure and deformation fields were coupled and solved simultaneously, thus FSI was taken into account in the analysis. However, the calculation was based on the acoustic elements, which ignore fluid flow. In the beginning the method was validated against the test cases calculated in Reference [1]. The pressure drop boundary condition was given in one cold leg. The pressure was observed at three points presented in Figure 64. The first point was located on the boundary condition surface, the second point on the surface of the inner basket next to the inlet pipe and the final point at the lower part of the inner basket. The pressure profiles are presented in Figure 64. The red curve (Node 13848) presents the boundary condition pressure at the pipe rupture surface. The pressure drops from 123 bars to 20 bars during 2 ms. The red curve (Node 13840) is the pressure on the surface of the inner basket next to the inlet pipe. The blue curve (Node 13411) shows the pressure at the lower end of the inner basket.

The boundary condition is physically reasonable, but the behaviour of the two other curves appear a bit surprising. Especially high subpressure levels inside the reactor and following oscillation amplitudes seems to be too large. Some kind of subpressure and oscillation would be reasonable because of movements of the inner basket, but the code is probably overestimating them. It looks like the applied calculation method, the acoustic elements without presence of the fluid flow, is not completely suitable for this case. At least the model must be tested better for example varying the time step. However, the conclusion is that the pressure field should be calculated by another method. The next step is to calculate the pressure field with a CFD code and transfer it into FEA code.

STAR-CD – ABAQUS calculation

The second step in the project has been linking of CFD and FEA codes. The used CFD code is STAR-CD and the FEA code is ABAQUS. In this work, the one way link between STAR-CD and ABAQUS was tested.

It was assumed that a large break occurs in one of the cold legs at the same position as in the above analysis. The breaking of the cold leg leads to a propagation of a large pressure transient into the downcomer. The core barrel and the core basket are subjected to large transient loads. The loads during the first ten milliseconds of the accident are analysed. The loads are used to analyse the integrity of the core barrel and the core basket during the accident. The pressure load on the core barrel is transferred to the ABAQUS FEA code by using the MpCCI library [2]. The walls are assumed to be rigid during the CFD simulation.

The CFD calculations were performed by using the commercial Star-CD version 3.15A code [3] which has previously been found to be suitable for calculating rapid pressure transients [1, 4]. Star-CD can handle pressure dependent density variations of liquids, which is necessary in modelling, for instance, water hammer or LBLOCA.

The present CFD model includes part of the downcomer in the vicinity of the cold legs. LBLOCA at one of the cold legs is modelled with a pressure boundary condition, which is obtained from one-dimensional simulation with the TMOC code [5]. It is assumed that at the position of the break, the pressure decreases linearly within 2 ms from 123 bars to 20 bars. A single-phase simulation of the LBLOCA is performed. It is assumed that the temperature is T_0 = 265°C and the initial pressure is 123 bars. The density of compressible water is initially ρ = 785 kg/m^3 and the speed of sound is v_a = 1.12 km/s, i.e., the pressure transient propagates 1.12 meters in one millisecond. It is assumed that the flow velocity is initially zero. Turbulence is modelled with the standard large Reynolds number k-ε model of Star-CD.

The mesh size in the azimuthal and vertical directions was about Δz = 110 mm. The total number of the grid cells was about 63000. The time step in the simulations was Δt = 2 μs. The length of the simulation was 10 ms, i.e., 5000 time steps. The numerical solution of the flow equations was performed with the PISO algorithm of Star-CD. In the spatial discretization of momentum and density, the monotone advection and reconstruction scheme (MARS) was used because it has previously been found to be accurate for rapid pressure transients. The Crank-Nicholson method was used for the temporal discretization. The simulation geometry and some results of the calculation are illustrated in Figure 64.

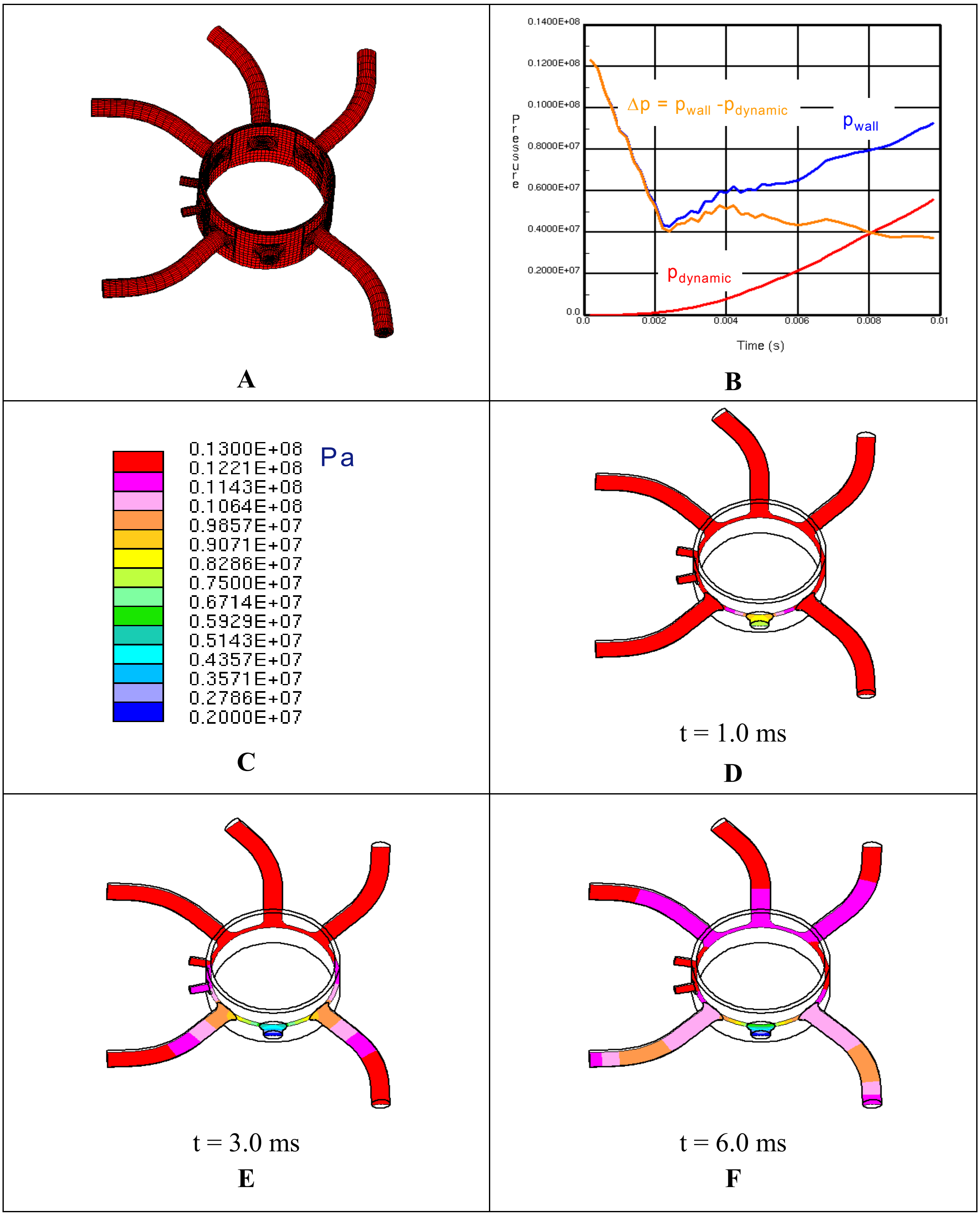

Figure 64. A: Simulation geometry, where the six cold legs of the pressure vessel can be seen.
B: Pressure (Pa) at the wall of the core barrel at the position of the broken cold leg. The wall pressure (blue line), the dynamic pressure at the outflow from the broken cold leg (red line) and their difference (orange line) are shown.
C–F: Pressure in a horizontal cross-section of the downcomer in the beginning of LBLOCA.

The pressure loads in the vicinity of the cold legs are illustrated in Figure 64 in the beginning of LBLOCA at three different instants of time. At time t = 1 ms, the pressure transient has propagated a distance of about one meter from the position of the break, where the pressure is still decreasing. At t = 3 ms, the pressure at the break position has already reached its final value of 20 bars. The pressure suppression front has propagated a distance of about three metres and has penetrated to the neighbouring cold legs. At t = 6 ms, the pressure transient has travelled in the azimuthal direction around the pressure vessel.

In the graph presented in Figure 64, the pressure is shown at the wall of the core barrel at the position where the minimum pressure occurs. In addition, the dynamic pressure is shown for the water flowing out from the broken cold leg. The difference of the wall pressure and the dynamic pressure remains approximately at the constant level of 40 bars after time t = 2.3 ms. This shows that the increasing wall pressure is caused by the dynamic pressure of the outflowing water. According to this model calculation, the pressure on the wall of the reactor barrel stays at about 40 bars when the pressure at the broken dead leg is 20 bars. The difference in static pressure in the nozzle region between the wall and the outflow is therefore about 20 bars.

After the CFD calculation, a relatively simple three-dimensional finite element (FE) model with geometrically and materially linear behaviour is used to analyse the response of the internals. The FE analyses are conducted with ABAQUS version 6.4.1. The analyses consist of an eigenvalue extraction and static and dynamic analyses with two different boundary conditions. In the static analyses, only a gravity load is applied to the structure. In the dynamic analyses, the gravity load and pressure load calculated in the CFD simulation are used. The model is presented in Figure 65.

Figure 65. Mesh of the FE model.

A four-node doubly curved general-purpose shell elements with finite membrane strains are used in the FE model. Every node has six degrees of freedom, three translational and three rotational. The size of the elements is on the average approximately 200 × 200 mm. The number of elements in the model is 2689, number of nodes 2785 and the total number of variables 16707. Different wall thicknesses are assigned to different regions of the FE model. Linear isotropic material properties are used with same values for each part of the model. Values E = 187 GPa, ν = 0.3 and ρ = 7850 kg/m^3 are used for Young's modulus, Poisson's ratio and density, respectively.

The analyses are divided into two different cases according to the applied boundary conditions. In both analysis cases translations of the upper edge of the structure are fixed while rotations of the edge are free. Also horizontal translations near the horizontal support level are fixed and all other degrees of freedom are free in both cases, i.e. vertical sliding of the structure is allowed at this location. In the second analyses case the gasket ring between the cold leg and hot leg connections is taken into account by fixing horizontal translations at the location of the gasket ring. The two different cases are calculated, since it is unclear how rigidly the gasket ring behaves in the analysed event.

The transient wall pressure is transferred from the CFD calculation to ABAQUS structural analyses code for the dynamic analyses. The pressures are interpolated from the CFD mesh to the FEA mesh due to the different meshing used in the CFD and FEA. The data file written by the interpolation program contains the transient pressure load on each element of the structural mesh on the coupling surface. The pressure load applied on each element is assumed as uniform over element face. In addition to the pressure load, a gravity load with gravitational constant g = 9.81 m/s^2 is applied in the dynamic analyses. In the static analyses, only the gravity load is applied.

Finally it can be said that the one way link between CFD and FEA is working well in principle. CFD results can be interpolated to FEA mesh and as a result useful information can be reached.

Conclusions

In this project studies of the FSI -problems using CFD and FEA codes have been carried out. The codes used in the work are NASTRAN, ABAQUS and STAR-CD. The application in this paper is a DBA situation in NPP. The specific case calculated is LBLOCA, which is a very important application from point of view of the Finnish nuclear industry, because it is a commonly analysed design basis accident for nuclear power plants.

The NASTRAN model was able to calculate the pressure transient, but the results seem to be at least partly little bit questionable. It is possible that the acoustic wave model used in NASTRAN calculations is not suitable for this case. There can be also some problems for example in meshing. This method should be studied more before further applications.

The STAR-CD – ABAQUS calculations were more successful. The pressure load was first analysed by a single phase STAR-CD calculation. The calculation was based on the pre-determined pressure boundary condition at the broken cold leg. The pressure was chosen to decrease linearly from 123 bars to 20 bars within 2 ms. The present analysis includes only the top part of the downcomer. The simulation region should be made larger so that the whole downcomer would be included in the model. A fairly coarse mesh with only 63000 grid cells was used in the present analysis. The mesh is not fine enough for producing the wall friction correctly when the flow velocity increases. The mesh is, however, fine enough for resolving the propagation of the pressure transient. Both the small simulation region and the underestimation of the wall friction may lead to too rapid acceleration of the outflow of water from the broken cold leg and to overestimation of the dynamic pressure caused by the outflow. The pressure loads were calculated by assuming rigid walls that do not move during the CFD simulation. Since the core barrel is expected move significantly under the pressure load, it would be of interest to repeat the analysis by taking into account the motion of the core barrel. This could decrease the pressure load on the core barrel. Even if the STAR-CD – ABAQUS calculation succeeded well, the results still need to be validated properly and the model development must be continued. The next phase in the project is to try to perform CFD – FEA -calculations using a two way coupling.

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10. APROS modelling of containment pressure suppression systems (TIFANY)

10.1 TIFANY summary report

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Abstract

This report summarizes the research and development activities performed in the TIFANY project during 2003 and 2004 through October. The aim of the project is to improve the capabilities of the APROS simulation program to serve as an integrated thermal hydraulic code for design basis accidents for different types of pressurized and boiling water reactors. In this report, the objectives and results of the project are described up to date.

The TIFANY project involves National Technology Agency of Finland, Fortum and Teollisuuden Voima as financing organizations, and Fortum Nuclear Services and VTT Processes as working organizations.

Introduction

The TIFANY project concentrates on improving the nuclear power plant (NPP) modeling capabilities of APROS simulation software. The main goal of the project is to improve the APROS containment calculation model to the level where diverse containment design basis accidents (DBA) for both pressurized water reactors (PWR's) and boiling water reactors (BWR's) can be reliably analyzed. As an example of a concrete objective of the project, Fortum is aiming at performing the Loviisa NPP containment licensing analyses with APROS starting from 2005. The project has also made improvements to other properties of APROS code to allow better modeling of advanced reactor types like EPR. The project also aims to widen the modeling possibilities of APROS to open new markets in Central and Eastern European NPP's.

The TIFANY project was first started in 2003 with Tekes[3], Fortum and Teollisuuden Voima as funding organizations and VTT Processes as a subcontractor. The good

[3] National Technology Agency of Finland

experiences gained in the project cooperation during 2003 lead to the idea of continuing the project in 2004 and 2005. The project has been running with a period of one year in order to make it possible to shift the focus of the research and development (R&D) to the areas, which have been identified as critical to the achievement of the overall objectives.

Main objectives

TIFANY 2003: The Integral Code for Design Basis Accident Analysis

Traditionally, the design basis accidents analyses for reactor cooling system and containment system have been separate, although the systems are thermal hydraulically connected during the accident. In the containment side, the analyses are usually based on blowdown curves calculated with the primary circuit model alone assuming that the containment is in a constant thermodynamic state.

The main objective of TIFANY in 2003 was to create and validate a generic integrated APROS calculation model for design basis accident analyses in various types of NPP's. A fully-integrated calculation approach is necessary for analyzing the behavior of various passive reactor and containment safety systems.

The lack of a simulation program integrating primary circuit thermal hydraulics and containment thermal hydraulics is an internationally recognized problem. Because of this lack, it was decided by VTT Processes and Fortum that such a feature should be added to APROS. As a project including this subtask and other APROS development subtasks TIFANY fitted well to the frameworks of SAFIR[4] and Tekes R&D program.

In addition to the development of the interface between APROS 5- and 6-equation model calculation and containment calculation, an objective was set to create generic integrated PWR and BWR plant application models including both a model of the primary circuit and the containment. The project is also aiming at validating recombiner and spray submodels of the APROS containment calculation and checking and correcting the correlations used in the APROS containment model and six-equation thermal hydraulic model to cover the needs of DBA analyses.

The TIFANY project in 2003 involved Tekes, Fortum and Teollisuuden Voima (TVO) as financing organizations and Fortum Nuclear Services (FNS) and VTT Processes as working organizations.

[4] The Finnish Research Programme on Nuclear Power Plant Safety

TIFANY 2004: APROS Modeling of Containment Pressure Suppression Systems

During TIFANY in 2003, a number of flaws and improper implementations of modeling were detected in the APROS containment model. Especially, improvements to the pressure suppression system modeling were considered to be necessary in order to achieve the level required for the containment licensing analyses. Based on the good experience of the TIFANY project in 2003, the project was decided to be continued with the same organizations involved.

The objective of TIFANY 2004 is to improve pressure suppression system modeling in APROS. Three topics were selected to the project: pressure suppression pool system, bubbler condenser modeling and ice condenser modeling. First one of these is of primarily interest in BWR's; second and third are interesting in special types of PWR's.

Modeling of pressure suppression pool is planned to be improved by introducing water pool temperature stratification model to APROS. The pool stratification model allows APROS water sump module to model two temperature layers in a water pool. Also, calculations of POOLEX pressure suppression pool experiments performed in Lappeenranta University of Technology are planned to be carried out to learn about the capabilities of APROS six-equation model capabilities and to test different condensation correlations.

Development of bubbler condenser model was included in the project because of its interest as a special technical solution for pressure suppression and to improve the marketing potential of APROS toward Central and Eastern European VVER-reactors utilizing this concept. The development and validation of bubbler condenser model is also improving the APROS containment calculation as a whole, since applying the code to various problems builds confidence to the capabilities of the code.

The aim of the APROS ice condenser model development is to implement the recognized features missing from the model, namely calculation of the free flow area, flow resistance and the temperature of the outflowing water.

In addition to these, a subproject involving the development of an advanced vertical steam generator model used in European Pressurized Reactor (EPR) was included in the project.

Main results

<u>TIFANY 2003: The Integral Code for Design Basis Accident Analysis</u>

The TIFANY project in 2003 was funded by Tekes and partly supervised by the Board of the SAFIR program, and some of the results of the project are confidential. TIFANY 2003 was finished in December 2003, and following results were gained from the project.

- An interface connecting the primary circuit thermal hydraulic model to the containment model was created and tested. The interface is currently a part of the APROS containment code. The use of the interface is documented in the APROS documentation [1].

- The APROS recombiner model was validated against data supplied by a well-known recombiner manufacturer. All APROS recombiner validation tests were successful (Table 9). The validation report is confidential [2].

Table 9. The results of the APROS recombiner validation tests. $\overline{\Delta c}$ is the average deviation of the calculated hydrogen concentration from the measured deviation during the experiment [2].

Test	Purpose	Result
RQT020	Recombination rate test, cold conditions Recombination rate test	$\overline{\Delta c}$ = 0.12 vol-% Recombination curve is OK
RQT025	Recombination rate test, hot conditions	$\overline{\Delta c}$ = 0.22 vol-%
RQT032	Recombination rate test, recombiner with chimney	$\overline{\Delta c}$ = 0.17 vol-%
Mass balance	Mass conservation test	Mass is conserved.
Energy balance	Energy conservation test	Energy produced by the recombiner is correctly transported to APROS thermal hydraulics.

- The APROS containment model was validated against Marviken test 18 representing typical BWR containment. The time history of the dry well and wet well pressures were well predicted. The calculated temperatures of the wet well atmosphere and pool differed from the measured values. The reason is assumed to be in too fast mixing of discharged steam in a control volume due to the lumped parameter approach. The validation report is confidential [3].

- The APROS containment internal spray system model was validated against the International Standard Problem no. 35 (ISP-35). The behavior of the pressure, temperatures and helium concentrations in the APROS test run were in good agreement with the measurements in the ISP-35 experiment. Suggestions as improvements to the spray model were made in the report and they are taken into account in TIFANY 2005.

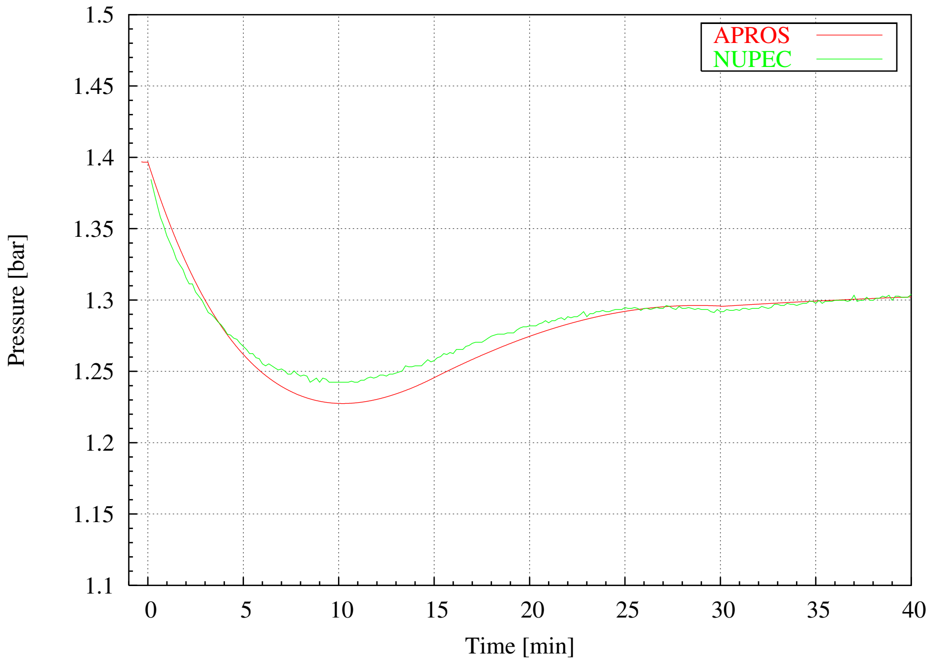

Figure 66. The measured and calculated pressure of the NUPEC test facility as a function of time in the ISP-35 experiment [4].

- A generic integrated PWR application model with a dry full pressure containment was created. The primary circuit model of the integrated model was tested and benchmarked against results of earlier APROS analyses. The full integrated model was tested by running several test cases with large LOCA and different number of ECCS trains available. The reports related to generic PWR model are confidential [5, 6].

- A generic integrated BWR application model with a containment was created. The model is based on the Olkiluoto plant. The model was validated against safety analysis codes GOBLIN and COPTA by running a steam line break simulation and against real plant data by running a reactor scram simulation. The reports related to generic BWR model are confidential [7, 8].

- The correlations used in the APROS containment and 6-equation thermal hydraulic models were checked and documented. Default input values were also documented for the containment model. The documentation is attached as a part of general APROS documentation [9, 10, 11].

TIFANY 2004: APROS Modeling of Containment Pressure Suppression Systems

The TIFANY project in 2004 is funded by Tekes and partly supervised by the Board of the SAFIR program, and some of the results of the project are confidential. The results of TIFANY 2004 will be ready in early 2005. Following results have been gained so far.

- A bubbler condenser model for APROS has been developed. The model is based on the geometry of Paks NPP in Hungary. The general behavior of the bubbler condenser system is currently validated against the results of EREC experiments. The report describing the bubbler condenser model is confidential [12].

- A new single node ice condenser model has been developed to APROS based on the ice condenser model used in VIRTA code. The model improves the calculation of flow friction and outflowing temperature in APROS. The new ice condenser model will be included in the next release version of APROS Containment. A draft of the reports is ready [13].

- An advanced vertical steam generator model has been developed to APROS. The model is tested in full power steady state conditions and with steam line break transient. The new steam generator model will be included in the next release version of APROS. The report describing the model is confidential [14], but the model will be documented in APROS documentation.

Applications

The improvements to the APROS calculation model are included in the official APROS releases available to all users with appropriate APROS license. The improvements are documented in the APROS documentation.

Several APROS primary circuit and containment model applications have been created in the project, e.g. Olkiluoto NPP plant and containment model, Paks NPP containment model and a model of the POOLEX-facility in Lappeenranta University of Technology.

Conclusions

The TIFANY project has started in 2003. The TIFANY project concentrates on improving the nuclear power plant (NPP) modeling capabilities of the APROS simulation software, especially in the containment calculation part. The project continues to the end of 2004, and continuation in the SAFIR program has been applied for 2005.

The project has produced many improvements to APROS and a number of APROS application models. The new features include a possibility to connect primary circuit thermal hydraulics and containment thermal hydraulics, several validated APROS calculation models and a number of new or improved process components. The improvements to the APROS calculation model are available to all users with appropriate APROS license. More results are expected in remaining of the project in 2004.

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11. Thermal hydraulic analysis of nuclear reactors (THEA)

11.1 THEA summary report

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Abstract

The project is focused on the thermal hydraulic analyses of nuclear power plants. Specific areas of research have been the modelling of heat transfer in horizontal steam generator in presence of non-condensable gas, and the development of tools for multidimensional two-phase flow simulations.

The effect of non-condensable gas on the heat transfer in the horizontal steam generator (SG) has been studied by calculating with APROS the PACTEL experiments NCG-1 (air injection) and NCG-3 (helium injection). The work done for the two-phase flow model development consists of two parts; improving the solution algorithm of porous media code PORFLO, and adding a homogeneous two-phase model to the commercial CFD code Fluent.

Introduction

To operate a nuclear power plant safely, it is essential to understand the thermal hydraulics and to be able to predict the plant behaviour during different transients. In predicting the plant behaviour, numerical simulation of thermal hydraulics plays a key role. The plant scale analyses are performed mainly with one-dimensional thermal hydraulic codes, e.g. APROS or RELAP. Such codes are indispensable in safety analyses, since they are the only way to predict the response of a large thermal hydraulic system, like reactor circuit. However, in many cases it would be necessary to be able to model the two- or three-dimensional features of the flow, which is often two-phase flow of water and steam.

CFD codes can capture the multidimensional effects, but they lack two-phase models suitable for nuclear applications. To overcome the lack of the desired models in the commercial codes, submodels can be added to the existing codes. If the exact modelling of geometrical details is not crucial, a practical approach is to use porous media model. In porous media codes, the vector fields are not solved in detail in the whole geometry, instead the effect of obstacles (such as tube bundles) is described through porosity of continuum. CFD and porous media models can be used as complementary methods; details of the flow field are solved with CFD codes, while larger entities can be analysed with porous media models.

A series of experiments have been performed in Lappeenranta University of Technology with PACTEL facility to investigate the effect of non-condensable gas on behaviour of heat transfer in horizontal tubes [3]. Though PACTEL is a model of VVER primary circuit, the effect of non-condensable gases is of general interest. While the effect of non-condensable gas in vertical heat transfer tubes is exhaustively investigated, the non-condensable gas behaviour in horizontal tubes is not yet fully understood. The behaviour can also be different depending on whether the gas is heavier than steam (air, nitrogen) or lighter than steam (helium, hydrogen).

Main objectives

PORFLO

The 3-dimensional porous media code PORFLO, has previously been developed at VTT. The code is based on PILEXP code, also developed at VTT, which has been used for simulating particle bed dryout experiments conducted at VTT [1, 2]. The PORFLO code uses five-equation two-phase flow model. In the model, mass and energy equations are written for both liquid and vapour phases, but the momentum equation is solved only for the two-phase mixture. The phase separation is solved using drift-flux model. In the first version of the code (same solver as in PILEXP), the solution of pressure and flow distribution was based on full matrix inversion. Due to the matrix inversion time, the problem size was limited to 20000 mesh points.

Objective of the PORFLO subtask was to realize the solution of the pressure and flow fields by using iterative algorithm belonging to the family of SIMPLE (Semi-Implicit Method for Pressure-Linked Equations) pressure corrector methods. This would enable solution of much larger problems, up to one million mesh points.

Homogeneous two-phase model in CFD simulations

Objective of the task was to implement a homogeneous two-phase model to Fluent code. This approach was chosen to be able to model evaporation and condensation, and to avoid the difficulties that arise when Eulerian two-phase model is applied to evaporation/condensation phenomena. In the homogeneous model, all properties like void fraction, density, temperature, viscosity etc. are defined as functions of pressure and enthalpy. There is no transport equation for the void fraction because the void fraction is determined as a function of pressure and enthalpy. This way all two-phase problematic matters have been included in the material property functions. In consequence, the CFD system has to solve just a single-phase fluid flow problem.

However, the homogeneous two-phase fluid has two difficult properties. First, the density can vary very much in the flow domain from pure steam (1 kg/m^3) to pure water (1000 kg/m^3) and this variation depends on the pressure (Figure 67). Second, the temperature-enthalpy relation is not invertible in the two-phase region (Figure 68).

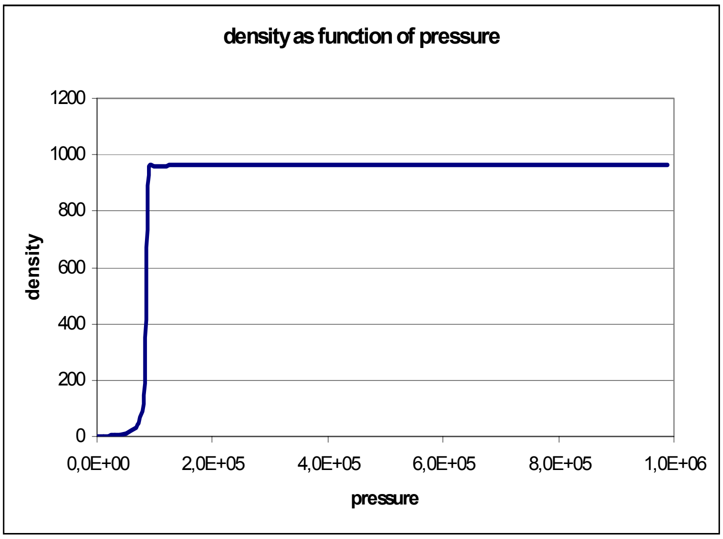

Figure 67. Density as a function of pressure for water.

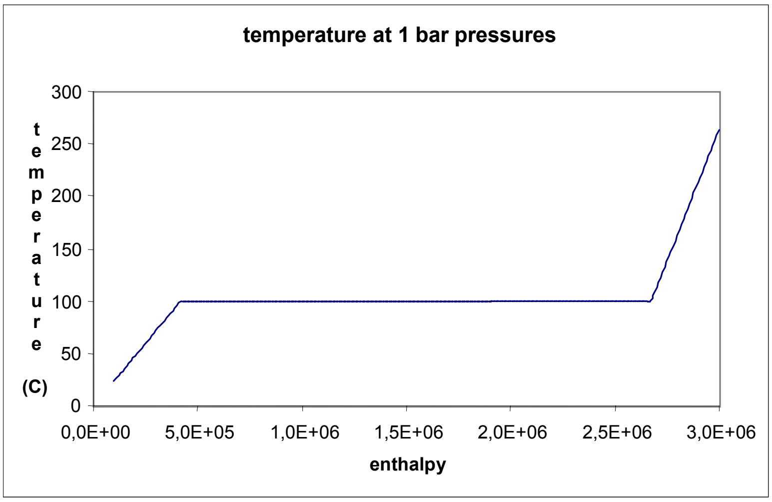

Figure 68. The temperature-enthalpy relation for water at 1 bar pressure.

APROS simulation of non-condensable gases in horizontal SG

Objective was to simulate with APROS the non-condensable gas tests performed with PACTEL facility. The experiments were made using two different non-condensable gases, air (heavier than steam) and helium (lighter than steam). The gases behaved clearly differently, but reasons for dissimilarities were not understood. It was hoped that the simulations would help to understand the effect of non-condensable gas on heat transfer in horizontal tubes.

Main results

PORFLO

Two alternative iteration methods (ADI – Altering Direction Iteration and SIMPLE-type algorithm) have been implemented to the code. The new code versions have been tested by simulating isolation condenser pool behaviour in the experiments performed at Lappeenranta University of Technology (Figure 69). Using either of the new methods, the simulation time is linearly proportional to the mesh number. With the original iteration method (broad band matrix inversion) the solution time was proportional to the 3rd power of the mesh point number.

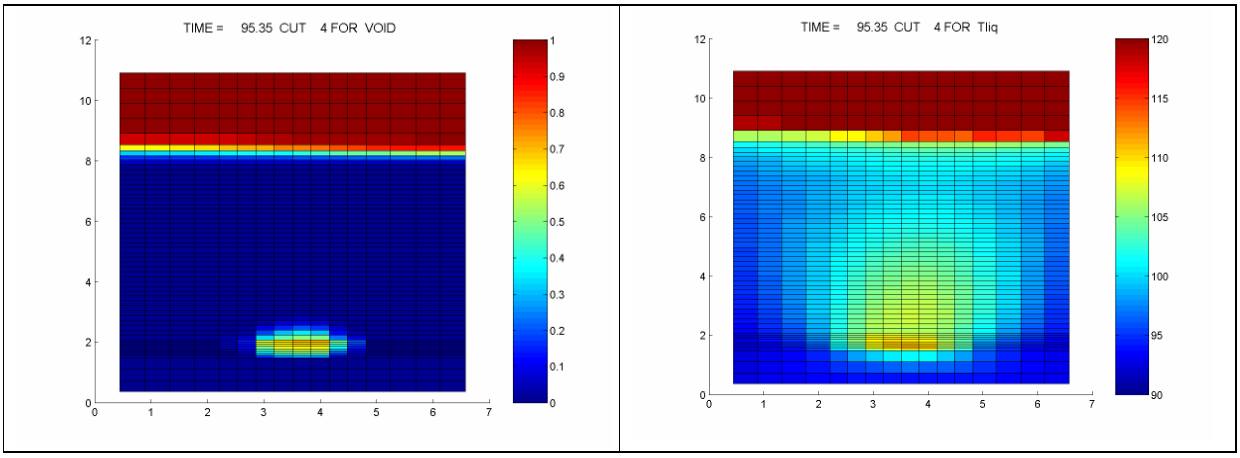

Figure 69. Void fraction (on the left) and temperature (on the right) distribution in the isolation condenser pool during initial phase of boiling calculated with PORFLO.

Homogeneous two-phase model in CFD simulations

The homogeneous two-phase model has been implemented in Fluent 6 environment.

The material property function and the fluid-wall heat transfer correlations have been coded. The problems related to the difficult temperature-enthalpy relation have been circumvented by replacing the standard enthalpy equation

$$\frac{\partial \rho h}{\partial t}+\frac{\partial (u_j h - F_{h,j})}{\partial x_j}=S \quad , F_{h,j}=k\frac{\partial T}{\partial x_j}, \ h=c_p T - c_p^o T^0$$

with the following enthalpy equation

$$\frac{\partial \rho h}{\partial t}+\frac{\partial (u_j h - F_{h,j})}{\partial x_j}=S\,,\ F_{h,j}=D\frac{\partial h}{\partial x_j}\ ,\ \mathrm{D} = \mathrm{D}_{lam} + \mathrm{D}_{turb}\ .$$

The diffusion coefficients D_{lam} and D_{turb} require further study.

Because it is not possible to define density as a function of pressure in Fluent 6 the compressibility effects on the flow field must be described by other means. As a first step the source term of the continuity equation was defined as

$$S=-\frac{\partial \rho'}{\partial t}-\frac{\partial (\rho' u_j)}{\partial x_j} \text{ where}$$

density change ρ' is due to pressure correction p'

$$\rho' = \frac{\partial \rho}{\partial p}p'.$$

So when continuity equation

$$\frac{\partial \rho}{\partial t}+\frac{\partial (\rho u_j)}{\partial x_j}=S$$

is used to solve pressure correction p' the source term contains information how density changes.

Unfortunately the term $\frac{\partial S}{\partial p'}$ is ignored in the Fluent solver. This is the key factor why no properly converging solution was attained in Fluent framework.

In SAFIR/INTELI project the experiences were put to use when implementing the homogeneous two-phase model in Star-CD environment. In Star-CD, better convergence has been achieved, and steady-state solutions have been attained.

APROS simulation of non-condensable gases in horizontal SG

PACTEL experiments NCG-1 (air injection) and NCG-3 (helium injection) were simulated. In the helium experiment the calculated heat transfer in presence of helium was too high, while in the presence of air the calculated heat transfer was too low. The parametric studies proved that in the calculation too much helium escaped from the SG while in simulation of the air experiment too much air stayed in the steam generator.

Conclusions

Two alternative iteration methods (ADI – Altering Direction Iteration and SIMPLE-type algorithm) have been implemented to the PORFLO code. The new code versions have been tested by simulating isolation condenser pool behavior in the experiments performed at Lappeenranta University of Technology. Using either of the new methods, the simulation time is linearly proportional to the mesh number, while with the original iteration method (broad band matrix inversion) the solution time was proportional to the 3rd power of the mesh point number.

No properly converging solution was attained in Fluent with the homogeneous two-phase model added. The problems were mainly due to the fact that it is not possible to define density as a function of pressure in Fluent. The model development was terminated when within SAFIR/INTELI, the homogeneous model was added to Star-CD with more success.

PACTEL experiments NCg-1 (air) and NCg-3 (helium) were simulated with APROS. Based on the simulation results, it became obvious that the accumulation of the lighter-than-steam and heavier-than-steam gases in the horizontal tube heat exchanger is different. APROS code assumes that steam and non-condensable gas form the homogeneous mixture, i.e. no separation takes place due to the mole weight of gases. Helium is lighter than steam, and therefore in reality it is collected to upper parts of the SG, while in the simulation too much helium is transported downwards with steam. On the other hand, air is heavier than steam and tends in reality to escape downwards more than the homogeneous calculation model indicates.

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11.2 Simulation of the PACTEL non-condensable experiments with APROS 5.04

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Abstract

This paper presents the comparison between the measured and calculated (APROS code version 5.04) results of the PACTEL experiment NCg-3. The experiment was carried out in natural circulation with only one loop in operation and at ~50% of the primary mass inventory. In the experiment NCg-3, compressed helium was used to simulate non-condensable gas. Non-condensable gas was injected in the vertical part of the hot leg below the steam generator.

APROS predicts qualitative trends of the test correctly, but some quantitative disagreements were observed. In the simulated experiment NCg-3, the primary pressure increases after the injection of helium, but not as much as in the test. Passive zones in the steam generator tubes seem to be smaller than in the test, indicating that heat transfer is too effective compared to the experimental results.

In the APROS calculation model it is assumed, that steam and non-condensable gas form a homogeneous mixture in the calculation node. APROS does not take into consideration the differences in the mole weights of the different substances. That increases non-condensable gas flow down wards from the steam generators cold collector. Helium does not accumulate in the steam generator as much as expected.

Introduction

Lappeenranta University of Technology (LTY) and VTT Energy (at present VTT Processes) have carried out experiments to study the behavior of non-condensable gases in VVER geometry [1, 2]. The tests aimed at studying the effect of non-condensable gases on the system thermal-hydraulics and on the heat transfer in a horizontal steam generator. The system behavior can be affected by hydrogen produced in the core in the case of severe accident, nitrogen from hydro-accumulators released into the primary circuit in case of a loss-of-coolant accident (LOCA) and more generally by any non-condensable gas in the all cases where cooling is ensured by natural circulation. The experiments can also help in validating the physical models and correlations that deal with non-condensables and that are implemented in thermal-hydraulic codes.

In particular, the presence of non-condensable gases could have a strong effect on the heat transfer in the horizontal steam generators of VVERs, where they would eventually accumulate. Non-condensable gases can greatly degrade the condensation process and reduce heat transfer efficiency. Another possible effect resulting from a large amount of non-condensable gas in the VVER geometry is the stagnation of natural circulation flow due to the loop seals.

Helium test and simulation

In the simulations, the standard six-equation model of APROS 5.04 was used. Small changes were made in the nodalization of the test rig compared to the original APROS nodalization of the PACTEL facility [3]. The steam generator model consists of the primary and secondary sides. The primary side has nodalization for the hot collector, for the horizontal heat transfer pipes and for the cold collector. The steam generator has been divided into five layers in the vertical direction. Each layer consists of the part of the hot and cold collectors, a number of heat exchange pipes and the corresponding secondary side layer. The pipes were divided into 20 nodes in the flow direction.

The third experiment of the series investigates the system response during a boiler-condenser mode natural circulation with helium as a non-condensable gas. Table 10 lists the event log for the experiment NCg-3.

Table 10. Event log for the experiment NCg-3 /1/.

Time [s]	Event
-11700	Facility heating starts, core power 75–200 kW, circuits 2 and 3 are closed.
-2700	Inventory reduction starts, circuit 1 pump is stopped, pressurizer is isolated.
0	Data recording starts, core power 170 kW.
1000	The first injection of helium starts, ~0.15 g/s.
1055	The first injection of helium stops.
1500	The second injection of helium starts, ~0.16 g/s.
1565	The second injection of helium stops, cumulative injected gas mass 16.5 g.
1895	The experiment is terminated, cladding temperature over 300ºC.

The examination of the pressure measurements shows that primary pressure increases after both injections of non-condensable gas (Figure 70). After both injections, a new steady state pressure level is achieved soon and equilibrium seems to prevail between the heat production in the core and the heat transfer to the secondary side and the heat losses. It can be found that the difference between the calculated and measured primary pressure becomes larger with the increasing helium content. After the first injection difference was 0.05 MPa and after the second injection 0.09 MPa. The secondary pressure was near to the atmospheric pressure all the time.

In the steady state situation, the measured pressure was very close to the measurement limit of the detector. It is very likely that the measured pressure was lower than the actual pressure in the system. The measured pressure does not correspond to the saturation temperature.

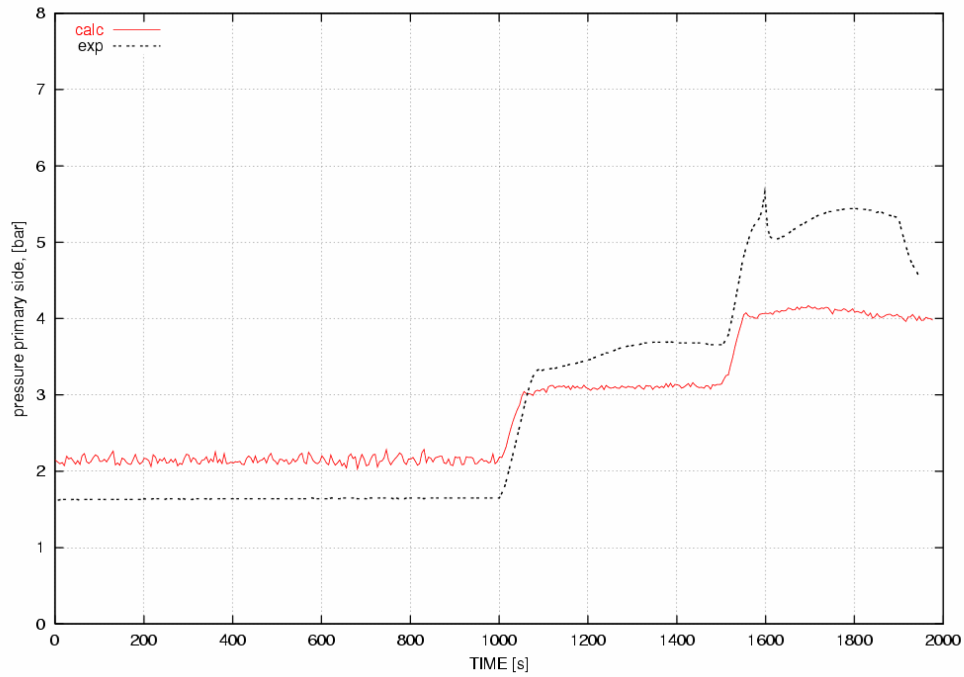

Figure 70. Calculated and measured primary side pressure.

Figure 71 shows the liquid mass flow in the cold leg. It confirms that the injection of helium causes a short stagnation of the primary flow during the injections. Calculated liquid mass flow is very unstable, but short stagnation period during the helium injection can be observed. The stagnation is much shorter than in the experiment.

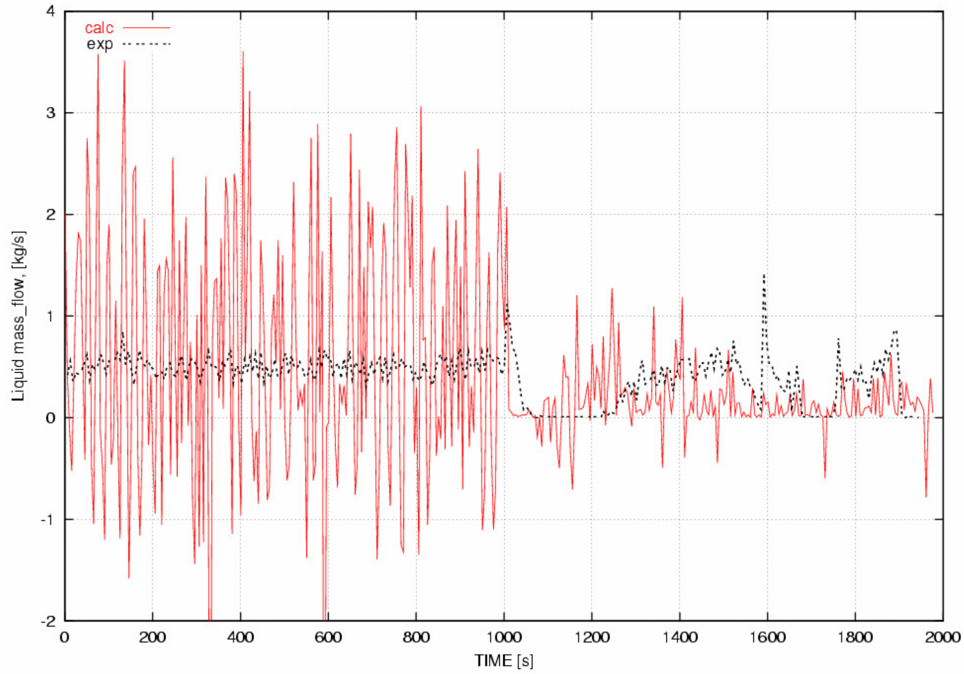

Figure 71. Measured and calculated liquid mass flow in primary side.

In calculations, some helium drifts with steam to the loop as can be observed in Figure 72. After the first injection most of the injected helium is located in the steam generator and about 13% in the circulation loop. After the second injection, more helium drifts to circulation loop and a small amount of helium can be found in the upper part of the downcomer. About 74% of the injected helium amount is located in the steam generator near the end of the transient. Helium drifts downward with steam flow in the calculations and drifting is probably too high in the calculations.

The density of helium is about 17% of the density of steam. In the APROS calculation model, it is assumed that the steam and helium form a homogeneous mixture in the calculation node. That causes increasing helium flow downwards from the steam generators cold collector. In reality the helium bubble, which squeezes out from the steam generator tube, probably starts to climb up when the flow velocities are low as in the current case.

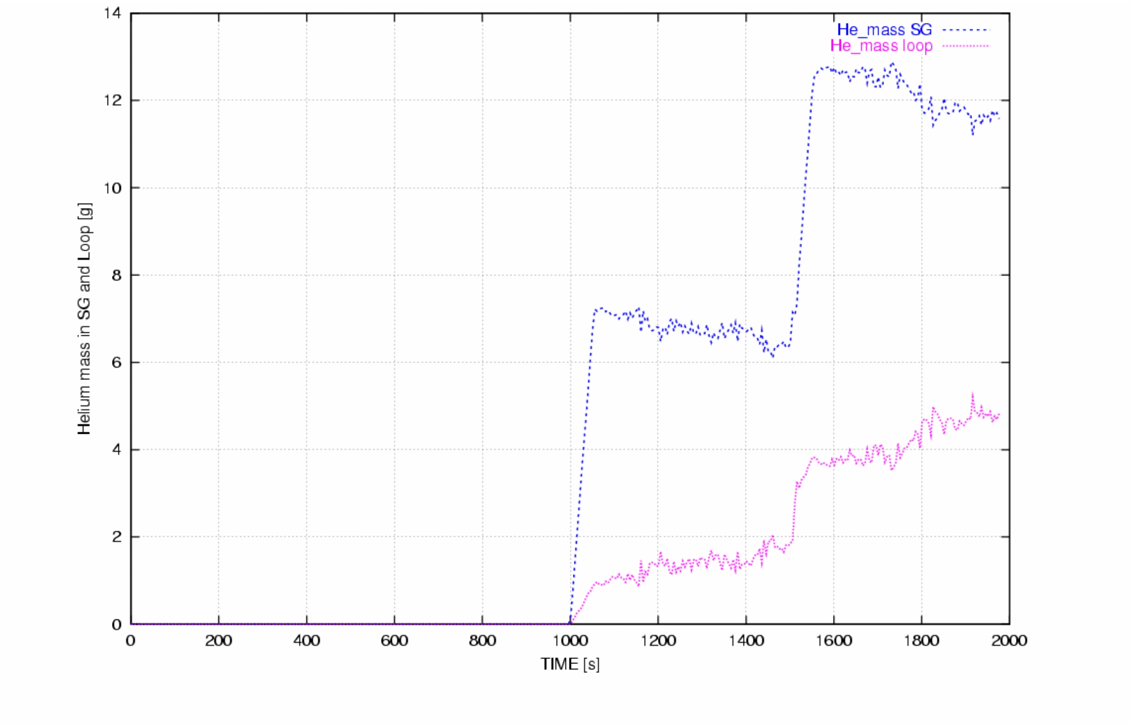

Figure 72. Calculated helium mass in the steam generator and in the loop.

It seems that the calculated helium distribution in the system differed from that of the experiment. This difference is believed to cause most of the differences in the steam generator heat transfer.

In the calculations, injected helium does not accumulate in the steam generator tubes as much as in the experiment. Therefore, the heat transfer is better than expected. It looks like that the calculated helium distribution differs from the test facility's distribution. A reason to this may be in the simulation model, where the gas mixture is treated as a homogeneous mixture.

Conclusion

PACTEL test with non-condensable gas helium injection was calculated with APROS 5.04. APROS predicts the qualitative trend of the test correctly. The primary pressure increases after each injection of non-condensable gas and passive zones in the steam generators tubes are formed.

In the helium case, the pressure and temperature increase is smaller than in the experiment. The quantitative differences are mainly consequences of too good heat transfer to the secondary side. In the simulation, a reason to this heat transfer problem is helium drifting out from the steam generator and too small passive zone in the steam generator tubes. The injected helium does not accumulate in the steam generator as much as expected.

The largest contribution to the efficiency of heat transfer in the steam generator is the amount of non-condensable gas in the tubes. The problem in APROS calculation is that helium does not accumulate in the steam generator. Because of homogeneous treatment of gas mixture in the six-equation model, helium flows with steam to the circulation loop.

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12. Severe accidents and nuclear containment integrity (SANCY)

12.1 SANCY summary report

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Abstract

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

Introduction

The objective of the SANCY-project is to reduce the remaining uncertainties of severe accident phenomena that are important to Finnish nuclear power plants. These issues comprise the core melt coolability, severe accident management in the long time range and updating of analytical tools to cover the also the needs of Olkiluoto 3 in addition to the existing plants. Furthermore, the follow-up and participation of major international research projects in the area of severe accidents is carried out as part of the project.

Furthermore, one of the goals of SANCY is to aid in maintaining competence in severe accident area in Finland by engaging experts from various fields of science and by providing education by doing for young students.

Main objectives

The STYX experiments aim at completion of concise data base on particular debris bed coolability in the containment for Finnish nuclear power plants, in particulr for Olkiluoto BWRs.

The long-term accident management studies will focus on assessing the survivability of containment penetration seal materials under high radiation doses, elevated temperatures and prevailing chemical environment.

The project aims at acquisition and implementation of new computer codes applicable also to Olkiluoto 3 plant. Also own computer codes will be developed utilising the information obtained from international experimental programs, in particular for melt coolability.

Project also follows-up the OECD/MCCI project investigating melt coolability and concrete erosion issues. Maintenance and updating of MELCOR code is performed in connection of participation to USNRC/CSARP programme.

Particle bed dryout heat flux experiments

The first test series was performed with a constrained (net on the top) stratified bed of alumina particles in the STYX test rig [1]. The purpose of the net on top of the fine layer was to limit the possible fluidisation of the fine layer. Measurements were performed at 2, 4 and 7 bar pressure. The difference between the constrained and open top of the fine sand layer was small though increasing at higher temperatures (Table 11). The dryout started near the bottom of the test bed.

The second test series was with a top layer of ferrochrome (FeCr) [2]. The density of ferrochrome is 1.6 times that of alumina. The effective grain size of FeCr was measured by Fortum NS to be 0.13–0.15 mm [3]. The sieved grain size of the fine sand was 0.2–0.4 mm.

The test bed was heated up to saturation and target pressure with two different methods: First by slow heatup with low power with the limitation of keeping the power level below dryout power during the pressurization and secondly by pressurizing the test section to target pressure with air which allowed the use of higher power for heating up of the bed to the saturation temperature. Steady state boiling was maintained for half an hour to remove air from the test vessel prior to start of step increases of power towards the dryout power. The measured dryout heat fluxes ranged from 215–365 kW/m2 (Table 11). Two different behaviour modes of dry-out were observed depending on initial pressurization method. Without initial pressurization with air the dryout was initiated near the top of the test bed. With initial pressurization with air the measured dryout powers were higher and the dryout started at a lower elevation in the bed.

Table 11. Dryout heat fluxes of stratified beds measured in STYX facility.

Pressure/Dryout power and heat flux	**2 bar**	**4 bar**	**7 bar**
No net, alumina top layer	205–221 kW/m^2	-	348–361 kW/m^2
Net on top of fine layer, alumina top layer	208–221 kW/m^2	294–307 kW/m^2	242–256 kW/m^2
Ferrochrome top layer, initial pressurization with air	215–240 kW/m^2	393–420 kW/m^2	325–337 kW/m^2

The third, ongoing test series, investigates the effects of bed height to dryout heat flux. The test will be performed with three separate test bed sections having heights of 200, 400 and 600 mm. The number of heater layers was increased to be 6 in 200 and 400 mm beds and 9 (instead of earlier 6) heater levels in 600 mm bed to obtain more uniformity in the heating. The first set of measurements with 400 mm homogeneous bed with alumina particles has been performed. The preliminary results indicate that a shallower particle bed has higher dryout heat flux (Figure 73).

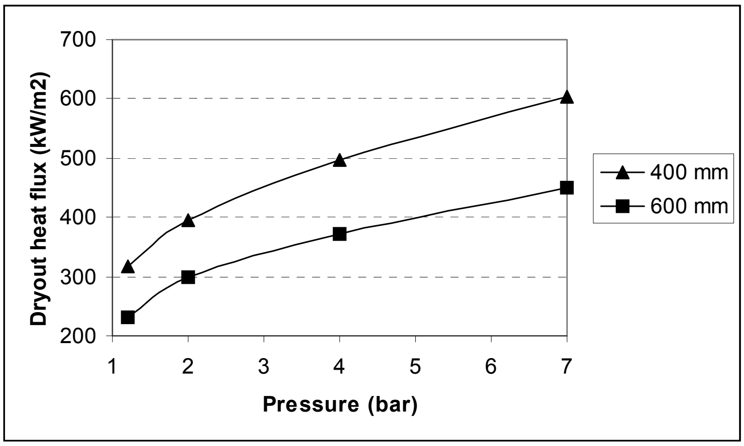

Figure 73. The measured dryout heat fluxes for 400 mm and 600 mm homogeneaous particle beds. The dryout heat flux increases as the bed depth decreases.

Seal material irradiation tests

Seal material behaviour under exposure to radiation during a long period of time after a severe accident is investigated with irradiation and material property testing of real seal material samples.

The first tested material was butyl rubber used in Olkiluoto 1 and 2 in personal hatch in the upper drywell and in service door in the transport shaft. The maximum cumulative dose to the butyl rubber samples was about 0.9 MGy corresponding to the estimated cumulative dose of walls during the first year after a hypothetical severe accident. The effects of seal material being in contact to caustic solution resulting from accident management has been accounted for in the testing. Also the period of elevated temperature in the upper drywell prior to completion of containment water filling has been accounted for in the testing. The material properties were significantly affected already with the shortest irradiation time corresponding to a cumulative dose in the containment after about 2 months from the accident.

A similar test series is under way using EPDM used as seal material for cable penetrations in Olkiluoto 1 and 2. The details of the butyl rubber and EPDM test results are presented by Zilliacus et al. in [4].

Development of water ingression model for ex-vessel debris coolability

The key mechanism for core melt pool cooling on the containment floor by top flooding has been recently recognized to be water ingression. Water ingression means a gradual penetration of water inside the melt by overcoming the counter-current steam flow in narrow cracks of solidifying melt resulting ultimately in cooling of the melt in excess to conduction limit through the solidified top crust. The importance of this cooling mechanism is particularly important to plants constructed of siliceaous concrete.

The water ingression has been investigated theoretically by Epstein in the ACEX project [5] and later experimentally with the SSWICS tests within OECD/MCCI project. Inspired by Epstein's model a computer code WATING modelling water ingression cooling was developed at VTT. The code has been applied to calculate SSWICS tests with promising results. More detailed discussion of water ingression modelling is presented by Kanerva in [6].

Adoption of GEMINI2/NUCLEA thermochemical database

The GEMINI2/NUCLEA code system was purchased from French company Thermodata and implemeted at VTT Processes [7]. The GEMINI2 code using NUCLEA database allows the user to calculate the phase diagrams of metal-oxide mixtures encountered in various in-vessel and ex-vessel core melts for further use in assessments of severe accident phenomena. The model estimates complex multiphase multicomponent chemical equilibria (ideal gaseous phase, stoichiometric condensed substances and multicomponent condensed solution phases) by minimization of the total Gibbs energy of the system under either constant pressure or volume conditions.

The code was used to determine the liquid fractions of a typical ex-vessel melt pools in PWR and BWR plants with varying mass fractions of dissolved concrete with composition used in Olkiluoto 1 and 2 (Figure 74).

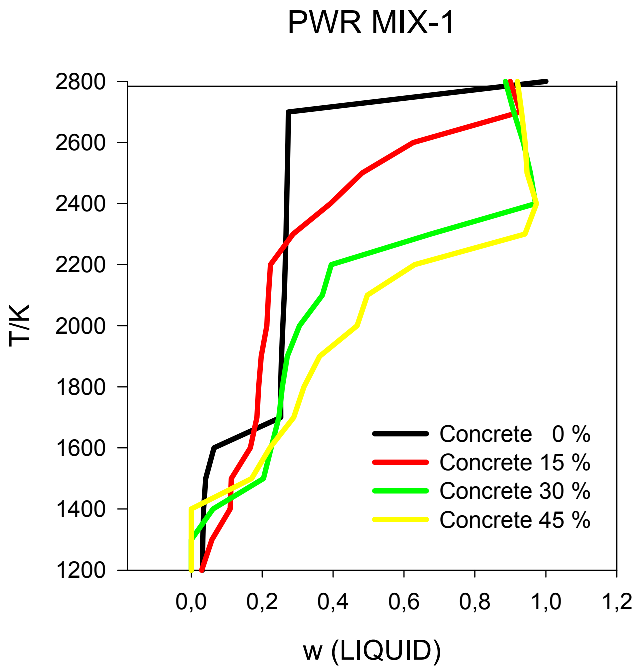

Figure 74. Illustration of melt mixture liquid fraction vs. interface temperature in a system of typical PWR core melt on ablating siliceous concrete floor [7].

Miscellaneous work and international follow-up

An assessment of remaining uncertainties of severe accidents was produced based on literature study [8]. Large uncertainties still exist in coolability of (ex-vessel) core melt pools and in long term effects of severe accidents. Uncertainties exist continuously in the fuel melting and relocation phenomena inside the pressure vessel. Some areas, e.g. physics and modelling of hydrogen combustion phenomena are currently rather well-known.

The recent 2-D core melt concrete erosion experiment CCI-1 was calculated with MELCOR 1.8.5 code. The study was enhanced with hand calculations on maximum achievable erosion depths and minimum decay heat removable via conduction from the melt used in the test [9]. The work was also an academic special assignment of a summer intern.

OECD/MCCI project performing tests on water ingression and melt eruption coolability mechanisms, solidified core melt (crust) structural strength and material properties and 2-D core melt-concrete interaction has been participated. Information of the obtained results and project progression have been realised in form of technical trip reports and technical data reports have been further distributed to STUK, TVO and Fortum.

USNRC/CSARP programme meetings have been participated. Latest versions of MELCOR code are obtained to Finland via participation to CSARP.

Applications

STYX particle bed experiments can be applied to assess the debris bed coolabilty in Olkiluoto BWR containment. The test data have been applied to model development. Seal material testing gives prototypical data applicable in assessment of long term containment leak tightness following a severe accident. Similar data is sparse or does not exist internationally. GEMINI2 has been already been used to other nuclear safety applications outside SANCY project. The acquisition, implementation and learning to use in SANCY project has facilitated timely start of other uses. WATING code will be further applied to estimate cooling behaviour of hypothetical melt pool in the Olkiluoto 1 and 2 containment.

Conclusions

Important new data on particle bed coolability have been obtained in the new STYX experiments. The dryout heat fluxes of homogeneous beds with wide particle size distribution may be lower than generally expected and on the other hand shallow, non-heated layers of fine particles may not reduce the dryout heat flux as much as predicted by the models.

New data have been produced on long-term behaviour of containment seal materials exposed to high radiation rates and chemical environment after a severe accident. This type of data is sparse or non-existing. In general the material properties of the seals are significantly changed already in a couple of months after a severe accident.

New calculation tools for assessments of severe accident phenomena have been adopted (GEMINI2) and developed (WATING).

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12.2 Study of water ingression models

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Abstract

It is important to understand the mechanisms of quenching molten corium material when estimating the safety of nuclear plants. The melt freezes creating an impermeable crust when water is injected on the top of the molten pool. The heat is carried away from the melt only by conduction through the crust and thus the cooling rate decreases as the crust grows. Cracking of the crust allows water to penetrate into the crust making conduction layer thinner and increasing the cooling rate. This is called water ingression.

The water ingression model presented here is partly based on experimental data obtained for rocks, and thus it is not clear if it can be used in the case of corium pools. If it is permissible to extrapolate this theory to relatively high-temperature corium, the water ingression is not possible with the corium pools. However, the theory is sensitive to errors with high temperatures. Experiments have shown that the cooling rate of the corium pools is higher than it should be with conduction only and careful analyses of experiments are required to verify the nature and scale of water ingression process.

Introduction

One important field of estimating the safety of nuclear plants is to understand the mechanisms of quenching molten corium material. The melt freezes resulting an impermeable crust over it when water is injected to the molten corium pool. The heat is carried out from the melt by conduction through the crust and thus the cooling rate decreases as the crust grows. However, the crust shrinks as it cools and this causes cracking of it. If the cracking is adequate, the water can penetrate into the crust so that the conduction zone becomes thinner and cooling becomes more efficient.

M. Epstein [1] constructed an unpublished steady-state thermal model of the process. In the model the solidification front is considered to be stationary and the molten corium moves upwards and solidifies when it passes the front. A thin layer of solid crust material is located above solidification front and on top of it is cracked crust. Water layer is located at the top of the crust and it penetrates to the cracks. The major obstacle to the prediction of the water ingression rate with Epstein's model is the absence of information on the permeability of the cracked crust.

C.R.B. Lister [2] wrote an extensive paper about penetration of water into hot rock and developed a theory for the penetration mechanism by considering simplest possible one-dimensional model. This is widely observed geophysical phenomena. Molten magma freezes when water comes into contact with it. The frozen magma shrinks and builds up stresses while cooling from above. The stresses are strong enough to crack the frozen magma and it results a crack matrix that propagates downwards. As part of this water penetration theory he developed an expression for the bulk permeability of the crack matrix.

Epstein coupled his water ingression model with Lister's permeability model. Because Lister's model relies to some extent on data obtained for specific rock material serious errors may arise when it is applied to relatively high temperature corium material. Nevertheless a feeling is obtained for the relationship between parameters when a crust cover growing on a highly convective corium pool is permeated by water.

Conceptual model for UO2 top cooling

Water ingression occurs when molten corium invades the floor of the containment and is quenched by injecting water to the top of molten corium. The molten corium begins to solidify from the top creating an impermeable crust layer (see Figure 75). The water can not pass through this solid layer and the heat is removed from the molten pool beneath it through the crust by conduction only. As the crust grows the top part of it cools down because of the water above it. The cooling crust shrinks and this causes stresses that crack the crust. If the cracks are extensive enough, then water can penetrate to the cracks. The boiling can then carry away heat deeper from the crust and the conduction layer becomes thinner. The cooling rate of the melt pool is thus increased. This water ingression might make it possible to cool down even dozens of centimeters molten, hot corium.

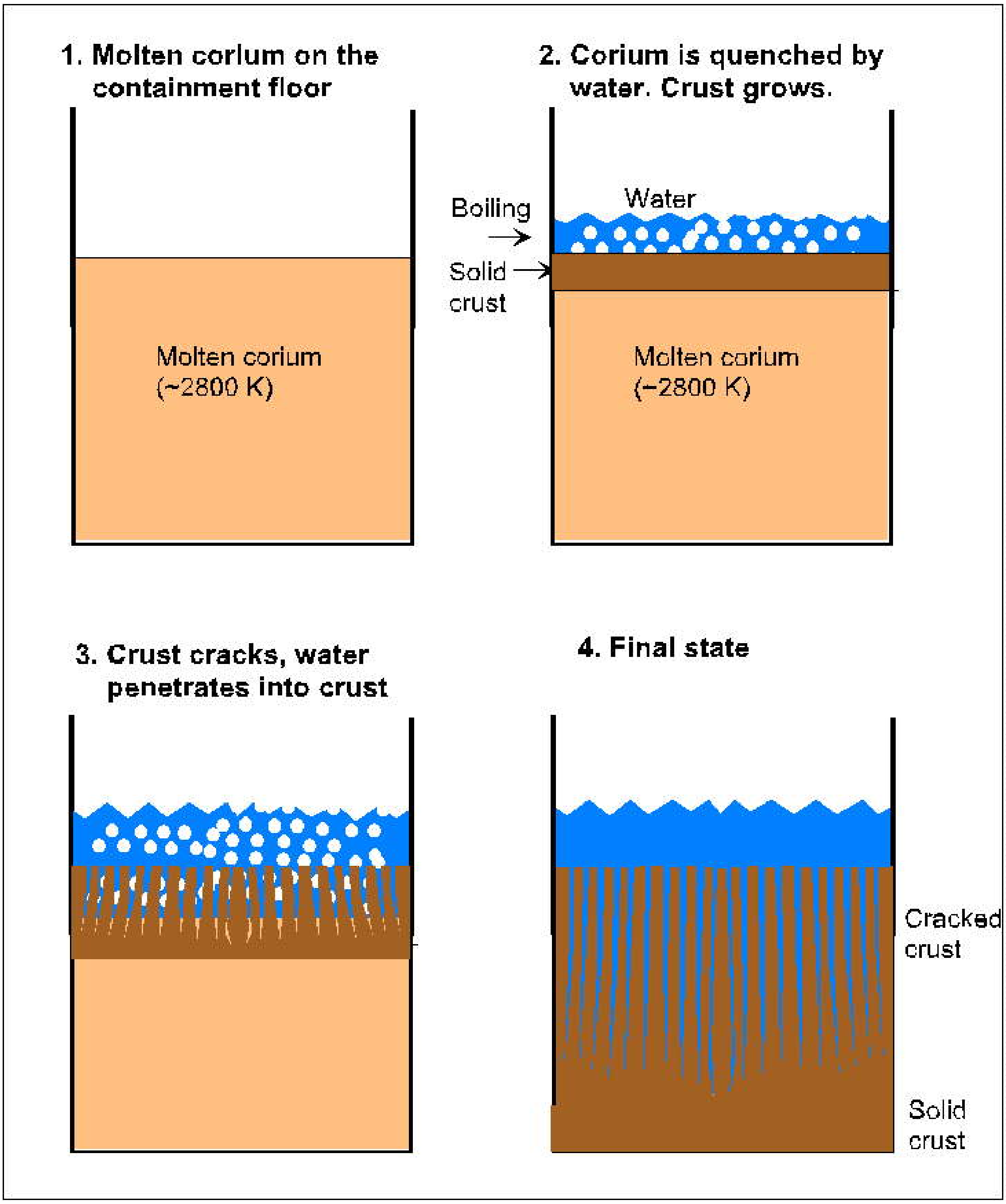

Figure 75. Solidification of molten corium pool with water ingression.

Epstein's water ingression model

In this Epstein's earlier model (1989) there occurs cracking in the crust above molten pool and the water penetrates into cracked crust (see Figure 76). The cracks do not traverse the entire crust and the water does not penetrate all the way to the crack tips. Upward heat flux from the melt pool equals the maximum heat transfer capacity of boiling water at some location $z = \delta$, where z is measured upward from the solidification front. This maximum or 'dryout' heat flux is denoted by q. Below that location there is a dry region $0 < z < \delta$ through which the heat flux from the pool is transmitted mainly by conduction. Since the heat is being removed at steady-state rate q, the solidification rate reaches a constant value. Thus the problem can be considered to be similar than conduction of heat in moving solids, where the interface of the melt pool and crust is assumed to be stationary and solid material passes through it with velocity u equal to the pool solidification velocity. The temperature distribution $T(z)$ in the dry conduction zone is then:

$$u\frac{dT}{dz} = \alpha\frac{d^2T}{dz^2} + \frac{Q}{\rho c} \tag{1}$$

where Q is the volumetric heat generation rate and ρ and c are the density and heat capacity of the frozen corium. At the crust-pool interface the temperature equals the melting point of the core material, T_{mp}, and the conduction heat flux is equal to the heat supplied by the heat-generating molten pool and the latent heat of fusion. At $z = \delta$ the temperature is the water saturation temperature, T_{bp}, and the temperature gradient is proportional to the dryout heat flux. Therefore the differential equation can be solved and with little approximation and some algebra equations for the crust growth rate and thickness of the dry conduction zone are obtained:

$$u = \frac{q - q_{up}}{\rho\left[h_{fs} + c(T_{mp} - T_{bp})\right]} \tag{2}$$

$$\delta = \frac{\alpha\rho\left[h_{fs} + c(T_{mp} - T_{bp})\right]}{q - q_{up}} \cdot \ln\frac{h_{fs} + c(T_{mp} - T_{bp})}{h_{fs} + (q_{up}/q)c(T_{mp} - T_{bp})} \tag{3}$$

where h_{fs} is the latent heat of fusion of corium material and q_{up} is the upward convective heat flux from the molten pool under the crust.

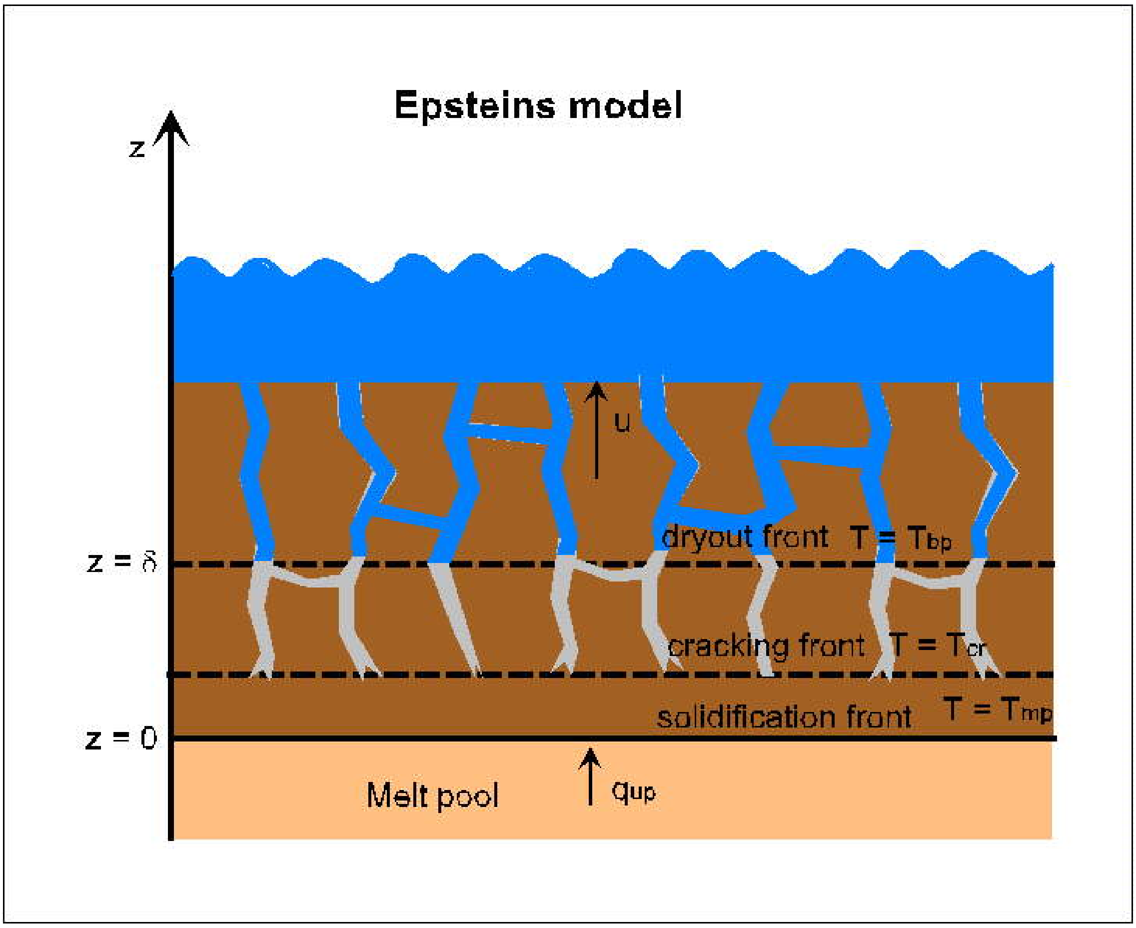

Figure 76. Visualization of Epstein's water ingression model.

The Epstein's model assumes that the boiling heat flux q is known. A reasonably accurate formula for the limiting heat flux q in a porous medium can be derived from Darcy's law:

$$q = \frac{h_{fg}\kappa(\rho_f - \rho_g)g}{2\nu_g} \tag{4}$$

where h_{fg} is the latent heat of evaporation of water, ρ_f and ρ_g are the water and steam densities, ν_g is the kinematic viscosity of steam and κ is the permeability of the porous medium. Actually the permeability of the crust is unknown and therefore the dryout heat flux is also unknown. Lister's model, however, permits an approximate relation between q and κ therefore giving a potential solution of the whole problem.

Lister's water penetration model

The area of interest on Lister's model of penetration of water into hot rock is his expression for the mean permeability κ of the cracked solid. A difficult treatment of the transient creep problem associated with lateral temperature gradients that cools the cracked crust is required to produce the permeability equation. Only main ideas of the model are presented here. Now in the solidified crust is a system of plane parallel cracks separated by spacing y. Permeability appears in Darcy's law and it can be related to crack diameter and crack spacing of the present problem. The crack diameter and crack spacing are connected via linear expansion of the crust material and temperature difference of cooling, and it remains to determine the crack spacing to obtain a predictive formula for κ.

Main idea in Lister's model is that dimensions of propagating crack system are controlled by radial temperature gradients. They cause hoop tension that subdivides the vertical solid columns between the cracks if the columns become too large. Lister assumed that the radial temperature drop ΔT_r over the cross section of a representative column of solid bounded by a polygonal crack pattern is comparable to the temperature difference required to generate the excess tensile stress that causes the cracking front to propagate.

The radial temperature difference ΔT_r is related to the vertical temperature drop $T_{mp} - T_{cr}$ equating the net rate of cooling $\dot{T}_{cr}$ at the cracking front with the conduction heat flux down the radial temperature gradient within the column. Approximating the radial temperature drop and estimating the local cooling rate with appropriate temperature profile gives finally equation for permeability κ

$$\kappa = \frac{\sqrt{2}}{12}\left[\frac{16\phi N\alpha^2}{u^2\left(T_{mp} - T_{cr} + \dfrac{h_{fs}}{c}\beta\right)}\right]^{4/5}\left[\alpha_T (T_{cr} - T_{bp})\right]^3 \tag{5}$$

where β is defined as

$$\beta = \frac{1 + \dfrac{q_{up}}{q}\dfrac{c(T_{mp} - T_{bp})}{h_{fs}}}{1 - \dfrac{q_{up}}{q}} \tag{6}$$

With these equations we can now estimate the dryout heat flux q and the water ingression rate.

Epstein's combined water ingression model and water ingression estimates

Although serious errors may arise when Lister's permeability model is applied to the relatively high temperature corium crust material, Epstein coupled it with his water ingression model at least to obtain a feeling for the relationship between parameters when corium pool is permeated by water. Equation (5) gives κ in terms of dryout heat flux q (via β, Eq. (6)) and pool solidification velocity u. Eq. (2) gives the relationship between u and q, and Eq. (4) between q and κ. We can eliminate κ and u by combining these equations and obtain following non-linear equation for q

$$q = \frac{h_{fg}(\rho_f - \rho_g)g}{2^{1/2} \cdot 12\nu_g} \left[\frac{16\phi N\alpha^2\rho^2 \left[h_{fs} + c(T_{mp} - T_{bp})\right]^2}{(q - q_{up})^2 (T_{mp} - T_{cr} + h_{fs}\beta / c)} \right]^{4/5} \left[\alpha_{\cdot T}(T_{cr} - T_{bp})\right]^3 \quad (7)$$

If the convective heat flux from the melt pool to the underside of the crust is negligible $q_{up} = 0$ and $\beta = 1$, and Eq. (7) can be solved explicitly for q

$$q_{q_{up}=0} = \left[\frac{h_{fg}(\rho_f - \rho_g)g}{2^{1/2} \cdot 12\nu_g} \right]^{5/13} \left[\frac{16\phi N\alpha^2\rho^2 \left[h_{fs} + c(T_{mp} - T_{bp})\right]^2}{T_{mp} - T_{cr} + h_{fs} / c} \right]^{4/13} \left[\alpha_{\cdot T}(T_{cr} - T_{bp})\right]^{15/13} \quad (8)$$

Epstein used Eq. (8) to estimate heat transfer from molten lava and compared theoretical result to measured value. Also front velocity, u from Eq. (2), thickness of the conduction zone, δ from Eq. (3) and other parameters for molten lava were compared to observations. All of these results appeared at least reasonable.

In situation with corium pools there is a significant upward convective heat flux q_{up} from the corium melt to the underside of the crust. When $q_{up} > q_{q_{up}=0}$, $q - q_{up}$ is small compared with q_{up} and terms involving $(q - q_{up})^2$ can be neglected. Setting also $q = q_{up}$ where q appears by itself in Eq. (7), we have explicit approximate relation

$$q - q_{up} = \left[\frac{h_{fg}(\rho_f - \rho_g)g}{2^{1/2} \cdot 12\nu_g q_{up}} \right]^{5/4} \left(\frac{16\phi N\alpha^2\rho^2 c}{q_{up}} \right) \left[h_{fs} + c(T_{mp} - T_{bp})\right] \left[\alpha_{\cdot T}(T_{cr} - T_{bp})\right]^{15/4} \quad (9)$$

If typical values for molten corium pool are inserted to this equation, results show that it would take several years to solidify the whole pool.

Conclusions

It is not clear if the water penetration theory developed by combining the models of Epstein and Lister can be used for making predictions for corium crusts. If it is permissible to extrapolate Lister's creep theory, based on data for rocks, to relatively high-temperature corium, calculation with typical molten corium pool values results that water ingression does not occur. However, the higher temperature makes the theory more sensitive to errors and also the values of the constants in the creep law may differ significantly from those for rock.

However, tests have shown that cooling rate of corium pool under a layer of water is higher than should be by conduction only. Therefore water ingression is likely to have notable effect on cooling process. Careful analyses of well controlled experiments are required to verify the nature and scale of water ingression process.

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12.3 The irradiation tests of sealing materials

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Abstract

Two different sealing materials used in the reactor containment buildings, butyl rubber IIR and ethylene-propylene-diene-terpolymer EPDM have been tested in the severe accident conditions, radiation field and simultaneous alkaline water exposure. There were clear differences between these two materials with radiation doses of some hundred kGy. Butyl rubber material is according to these experiments more sensitive for the radiation than EPDM, and already 115 kGy dose is changing the hardness values of the material considerably. EPDM is obviously undamaged after irradiation with 260 kGy dose.

Introduction

The personnel hatch in the containment building at Olkiluoto 1 and 2 have seals of IIR. A literature study carried out by Fortum NS revealed that a radiation dose 890 kGy might cause considerable weakening of the seals. In Olkiluoto, the personnel hatch in the lower dry well comes in to contact with alkali solution immediately upon flooding of the lower dry well. The personnel hatch in the upper dry well may later be submerged with water, when the flooding is performed. The service door in the transport shaft is in the air in the upper dry well. The radiation dose calculations made in the MOSES project gave a 1 MSv cumulative dose for the lower dry well and wet well water pools and also for the air in the upper dry well within a year of a severe accident [1] PRO1/T7025/02/. Sandia reports of experiments with doses 200 times greater than those calculated in MOSES, and with a high dose rate [2].

In experiments performed 2003 the seals butyl rubber IIR used in Olkiluoto was tested in the SANCY project. Gammacell ^{60} Co- source of VTT Processes was used for the irradiations. The radiation dose rate of the experiments was 340 Gy/h. The longest irradiation went on for 16 weeks corresponding to a total dose of 900 kGy. In 2004 EPDM sealing material was used in similar tests.

The seal properties after irradiation were studied in VTT Processes, Materials and chemicals/Tampere. The aim of the study was to check the influence of irradiation on these two materials. For that purpose, the tests chosen were tensile strength, Dynamic Mechanical Thermal Analysis (DMTA) and Shore A hardness tests.

Irradiation tests

The samples were die-cut into a standardised dum-bell form in order to get suitable samples for tensile strength and DMTA measurements (Figure 77).

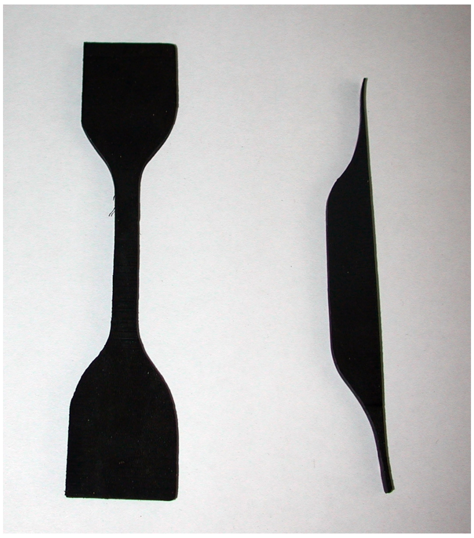


Figure 77. Samples for irradiation tests.

The samples were the tested in three series:

1. Irradiation in air without any pre-treatment (A)
2. Irradiation in alkali solution without any pre-treatment (B)
3. Irradiation in alkali solution after heating in 100°C for 100 h (C).

Each set of samples contained five parallel samples for tensile strength tests and two parallel samples for DMTA tests. The sample codes, doses and procedures are given in Table 12.

Table 12. Sample codes, doses and sample treatment.

Sample code	**Irradiation time, weeks**	**Dose, kGy**	**Dry irradiation**	**Irradiation in alkali solution**	**Heating in 100°C**
115A	2	115	X		
115 B	2	115		X	
115 C	2	115		X	X
340 A	6	340	X		
340 B	6	340		X	
340 C	6	340		X	X
900 A	16	900	X		
900 B	16	900		X	
900 C	16	900		X	X
96 A*	2	96	X		
96 B*	2	96		X	
96 C*	2	96		X	X
270 A*	5	270	X		
270 B*	5	270		X	
270 C*	5	270		X	X

* 2004 samples of EPDM sealing material

The alkali solution used in these experiments was NaOH (0.0001 M), corresponding to the one calculated for the containment water in Olkiluoto 1 and 2 after flooding/NKS-42/.

Tensile strength test

The tensile strength tests were done using Instron Model 4500-equipment following the rubber/elastomer standard without extension meters for samples irradiated with doses of 96, 115, 260 and 340 kGy. The samples of IIR irradiated with a dose of 900 kGy were so badly damaged that no tests were possible. The EPDM samples with total dose of 900 kGy are still in the irradiation and will be tested next year. The untreated reference samples were tested simultaneously with the irradiated samples (marked as reference in the tables).

The results of the tests are presented in the Table 13.

Table 13. Results of the tensile strength tests.

	Max Load (N)	Stress at Max Load (MPa)	Strain at Max Load (%)
Reference IIR	148	9.43	499
115 kGy A	101	6.30	358
115 kGy B	115	6.95	411
115 kGy C	102	6.09	429
340 kGy A	17	1.04	246
340 kGy B	36	2.19	288
340 kGy C	32	1.98	298
Reference EPDM*	121	7.23	612
96 kGy A*	104	6.33	564
96 kGy B*	116	6.98	478
96 kGy C*	135	8.13	345
260 kGy A*	89	5.39	486
260 kGy B*	108	6.56	383
260 kGy C*	127	7.72	283

DMTA measurements

DMTA, Dynamic Mechanic Thermal Analyses is a method, for measurement of dynamic modulus and damping over a wide range of temperatures and different frequencies. From the measured data E´ and E´´, Tg (glass transition point) can be calculated. Tg values are characteristic to each and depend on the microstructure of elastomers. The samples were analysed in tensile mode over a temperature range of -100–100°C.

E´ (storage modulus) and Tg, glass transition point (tan d)

E´ is shown as a function of temperature for both materials in samples C in Figures 78 and 79. The E´ modulus describes the stiffness of the specimen. The maximum of tan d is the glass transition point, Tg. The results for samples A and B were similar with these results.

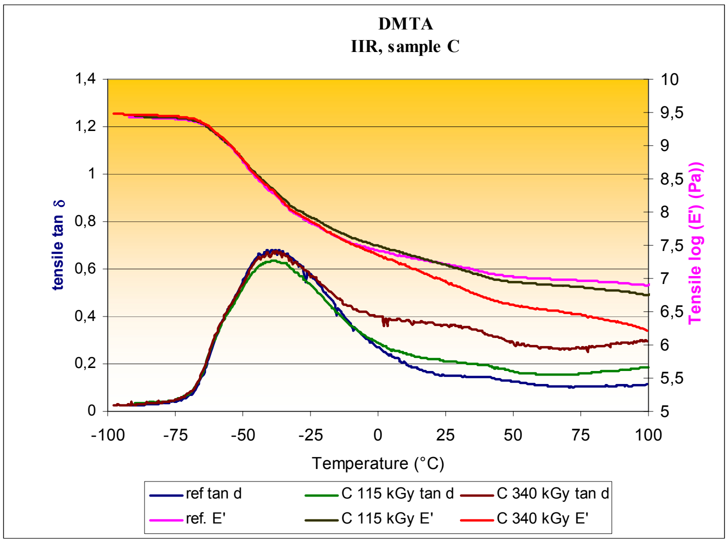


Figure 78. DMTA analysis of IIR, sample C.

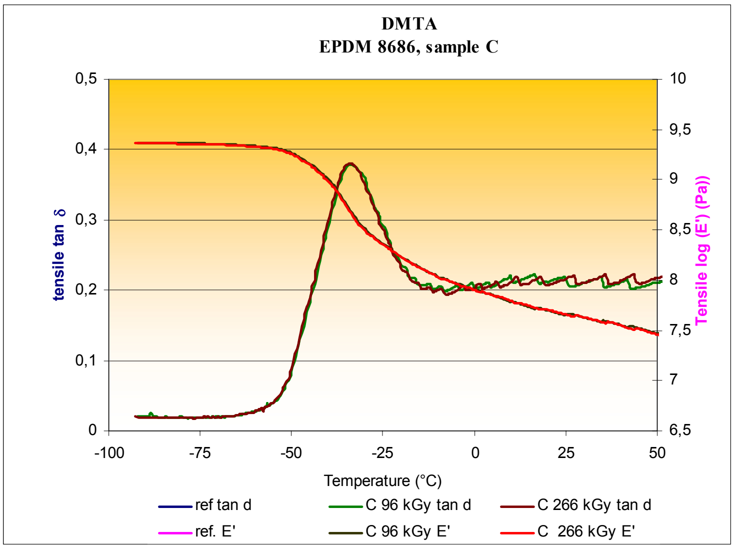


Figure 79. DMTA analysis of EPDM, sample C.

Shore A-hardness

The Shore A-hardness test was carried out with the durometer type A. The test method used for determining the relative hardness of thermoplastic elastomeric, rubber, and plastic materials is ASTM D 2240. This test method is based on the penetration of a specific indenter into the material under controlled conditions (force, material thickness, time to read etc.).

Butyl rubber IIR

The test results for IIR are presented in the in Figure 80 (the higher the value the harder the material).

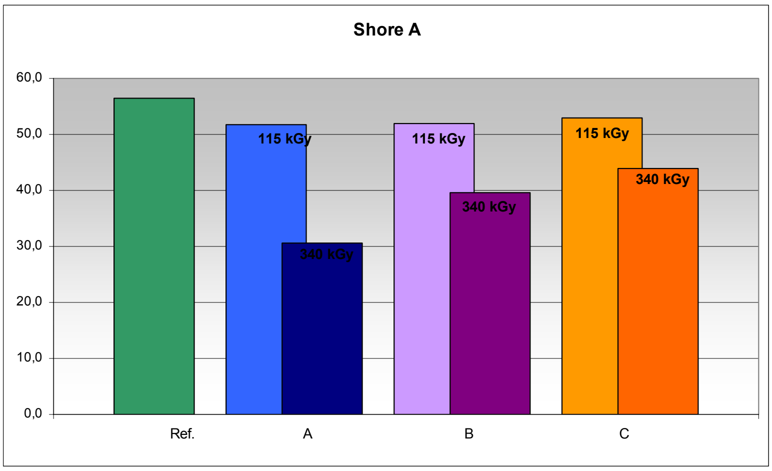
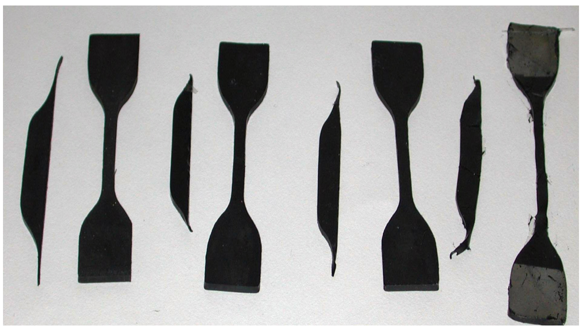


Figure 80. Results of the hardness test for IIR.

All the values become smaller than reference as soon as after two weeks of irradiation with a 115 kGy dose. It is clearly seen, that the double bonds of IIR elastomer are breaking due to the radiation and heating. This means that the elastomer is becoming softer. Irradiation in air instead of water seems to be more damaging.

Figure 81 shows samples irradiated with a total dose of 900 kGy for 16 weeks. The material in all samples was damaged so badly that it was not possible to do any tests. The samples irradiated in alkali solution kept their form better than those irradiated in the air.



Reference 900C 900B 900A

Figure 81. Butyl rubber samples after 16 weeks of irradiation with 900 kGy radiation dose.

According to these tests the butyl rubber sealing material is degrading with a dose of 115 kGy and is thus not suitable for sealing purpose any more.

EPDM samples

Figure 82 shows the results of the hardness tests for EPDM sealing material. According these results, EPDM, is standing exposure of radiation and heat. EPDM still have the elastic properties after exposure of 260 kGy and is obviously undamaged.

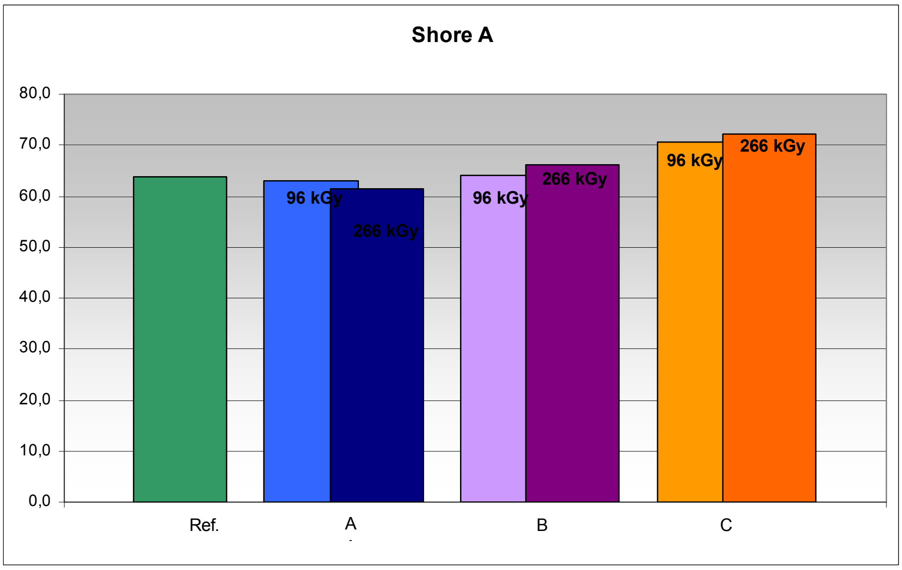


Figure 82. Results of the hardness test for EDPM.

Conclusions

Comparing the resistance of irradiation and heat of these two elastomer, butyl rubber (IIR) and ethylene-propylene-diene terpolymer (EPDM), it seems to be quite clear that EPDM stand up better. It is mentioned also in literature [3]. The reward of EPDM is good weathering and temperature resistance.

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13. Fission product gas and aerosol particle control (FIKSU)

13.1 Ruthenium behaviour in severe accident condition

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Abstract

In this project the transport and speciation or ruthenium in conditions simulating an air ingress accident was studied. Ruthenium dioxide was exposed to oxidising environment at high temperatures (>1200°C) in a tubular flow furnace. At these conditions volatile ruthenium species were formed. A major part of the released ruthenium was deposited in the tube as RuO_2. Depending on the experimental conditions 12–35 wt-% of the released ruthenium was trapped in the outlet filter as RuO_2 particles. At completely dry conditions using stainless steel tubes only 0.1–0.2 wt-% of the released ruthenium reached the trapping bottle as gaseous RuO_4. However, when alumina was applied as tube material or the atmosphere contained some water vapour the fraction of RuO_4 reaching the trapping bottle increased to 5 wt-%, which is close to thermodynamic equilibrium at the temperature of the furnace. This indicates that RuO_4 was not decomposed in these experiments.

Introduction

During the operation of a nuclear reactor, ruthenium will accumulate in the fuel in relatively high concentrations. In steam atmosphere ruthenium is not volatile and only a small fraction of it is likely to be released from the fuel, even if the fuel melts [1]. However, in an air ingress accident, ruthenium may form volatile oxides, which may be released to the containment. As the radiotoxicity of ruthenium is high in both short and long term, the understanding of its transport and speciation is of primary importance in case of an air ingress accident.

When RuO_2 is exposed to oxygen at high temperature, it reacts to form RuO_3 and RuO_4. At temperatures below approximately 700°C RuO_3 becomes thermodynamically unstable and decomposes to RuO_2. RuO_4 does not necessarily decompose upon cooling as it is metastable. It has been found to exist in appreciable amount at ambient temperature [2]. It can be noted that the reaction rates of ruthenium compounds are generally slow [3].

Thermodynamic calculations

Thermodynamic equilibrium calculations for ruthenium species in air and in 50 wt-% air-steam mixture were carried out using ChemSage5.0 software [4]. The result of the calculation for air is presented in Figure 83a. From the figure we can see that the most important vapour species at temperatures below 1300°C are RuO_3 and RuO_4. Above that temperature gaseous RuO_2 is also formed in larger quantities. At temperatures above about 1450°C decomposition of ruthenium oxides to metallic ruthenium starts to decrease the fraction of gaseous ruthenium species. In Figure 83b the results from the calculations for 50 wt-% air-steam mixture are presented. The results are qualitatively rather similar to the case with air, except that an additional gaseous ruthenium compound, RuO_3OH, is formed. Also the decomposition of ruthenium oxides to metallic Ru starts at a lower temperature.

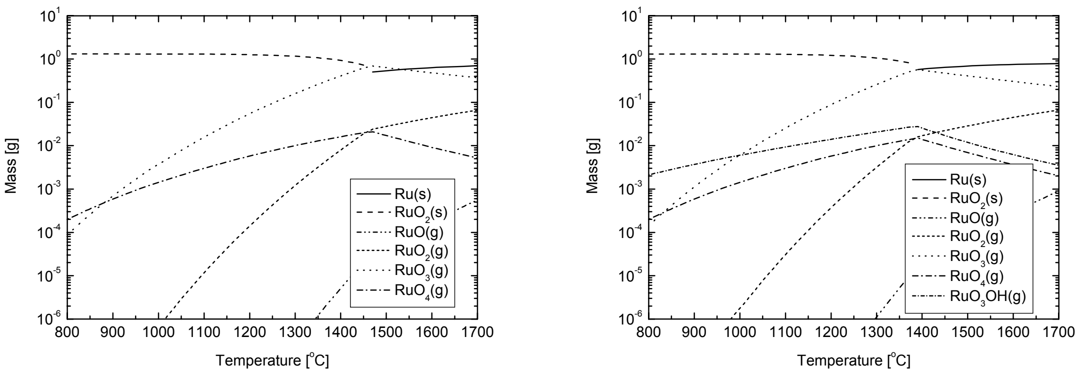

Figure 83. a) Ruthenium species at thermodynamic equilibrium in air at 1 bar pressure. b) Ruthenium species at thermodynamic equilibrium in 50 wt-% air-steam mixture.

Experimental

The laboratory facility used for the experiments on ruthenium behaviour is schematically presented in Figure 84. A tubular flow furnace was used to heat the ruthenium source (RuO_2 powder). The furnace tube had an inner diameter of 22 mm and it was made of alumina (99.7%). A 20 cm long alumina crucible with RuO_2 powder was placed 25 cm from the outlet of the furnace. During an experiment the gas flow rate through the furnace was 5 l/min. The release rate of ruthenium from the crucible in the furnace was determined gravimetrically from the mass loss of ruthenium dioxide during the experiment.

Gas-phase sampling of the particles was done 74 cm downstream of the furnace. The sample was diluted with a porous tube diluter in order to minimize the losses. For electron microscopy analysis particles produced in the system were collected on carbon coated copper grids using an electrostatic precipitator (ESP). In the transmission electron microscopy (TEM) analysis the size and morphology of the particles were studied. Selected area electron diffraction analysis (SAED) was carried out with the microscope as well. The microscope used in these studies was a Philips CM-200 FEG/STEM operated at 200 kV. Also the number size distribution of the particles was measured from the sampling line with a differential mobility analyser (DMA, TSI 3081) coupled with a condensation nucleus counter (CNC, TSI 3022). Before entering the DMA the particles pass through a pre-impactor, which removes larger particles. Small particles are then size classified according to their electrical mobility by the DMA. Particle number size distribution is acquired as the CNC counts the number of particles in each size class.

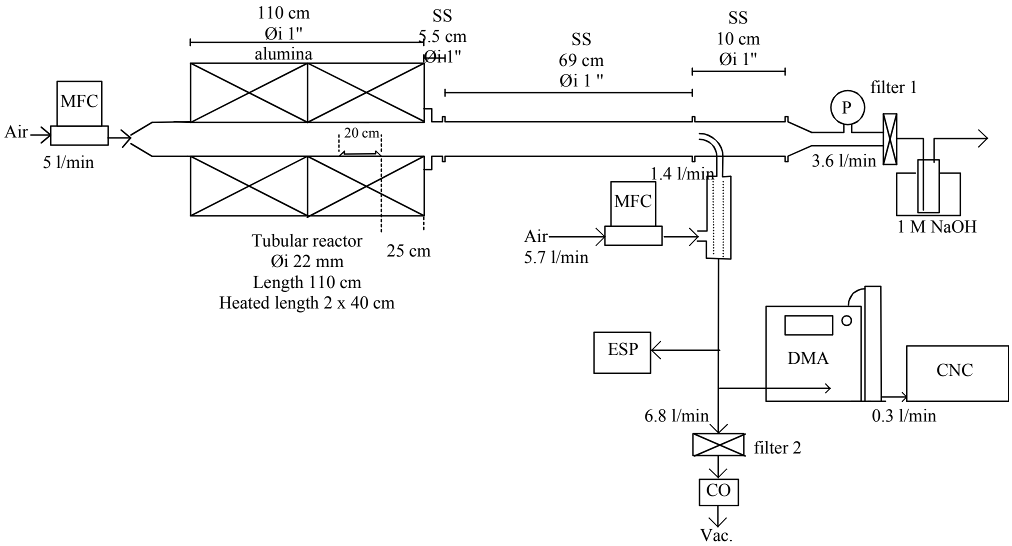

Figure 84. Schematics of the facility used in the experiments. MFC – mass flow controller, SS – stainless steel, ESP – electrostatic precipitator, CO – critical orifice, DMA – differential mobility analyser, CNC – condensation nucleus counter.

In the main flow channel RuO_2 particles were filtered out from the gas flow 106 cm downstream of the furnace. At this point the temperature of the gas was about 50°C. Filters from the first experiment were crushed and HCl and water was added in order to dissolve RuO_2 particles. After that the solution was analysed with ICP-MS. However, using this method all ruthenium could not be analysed as some ruthenium remained in the precipitate. Precipitates were clearly visible in the fluid. In later experiments the filters were analysed with INAA.

Gaseous ruthenium was trapped downstream of the particle filter into a 1 M NaOH-water solution. The trapping solutions were heated on a sand bath and ethanol was added to reduce ruthenates to RuO_2. In the first two experiments the precipitate from the trapping solution was analysed with ICP-MS (inductively coupled plasma mass spectrometry). In later experiments the precipitate from the trapping solutions were filtered and the filters were analysed with INAA (instrumental neutron activation analysis).

The experimental matrix with details on the experiments is presented in Table 14. Altogether six experiments were carried out. In the experiments the effect of release temperature, tube material, seed particles and oxygen partial pressure was studied.

Table 14. Details on experiments carried out on ruthenium behaviour.

Exp #	$T_{release}$ [°C]	Gas (5 l/min)	Tube material	Other	Main line filter	Sample line filter	Duration [min]
1	1227	Air	Stainless steel		Balston quartz	Balston quartz	60
2	1227	Air	**Alumina**		Nuclepore	Nuclepore	45
3	1227	Air	Stainless steel	**Tracer used**	Nuclepore		42
4	**1427**	Air	Stainless steel		PTFE	Nuclepore	20
5	1227	**N_2 90% O_2 10%**	Stainless steel		PTFE	Nuclepore	60
6	1227	Air	Stainless steel	**Seed particles**	PTFE	Nuclepore	38

Results

The results from the experiments are summarised in Table 15, where the release rate and the mass flow rates of gaseous ruthenium and ruthenium aerosol is presented. It was assumed that the release rate was constant and that ruthenium was released only when the gas flow was on. The mass flow rate for RuO_4 and RuO_2 in the trapping bottle and in the filter, respectively, are the mass flow rates of ruthenium, not of the oxides. The results are normalised to a flow rate of 5 l/min (NTP), because the carrier gas flow rate through the main line filter and the trapping bottle was not, due to sampling, the same in all experiments. As explained previously in the experimental section, the result for the RuO_2 aerosol mass in experiment #1 is not correct.

From Table 15 we can see that the release rate of ruthenium was almost constant (9–11 mg/min), when the release temperature and the gas composition were constant. If the system is assumed to be in thermodynamic equilibrium, 91% of the ruthenium would have been released as RuO_3 at 1227°C and 8% as RuO_4. Higher furnace temperature increased the release rate to 25 mg/min. At this temperature 94% of the released ruthenium was RuO_3 and 3% was RuO_4 according to thermodynamic equilibrium. Lower oxygen concentration (10%) slightly decreased the release rate to 6.6 mg/min.

Table 15. Summary of results from ruthenium experiments.

	Release rate [mg/min]	**RuO_4 In trapping bottle [mg/min]**	**RuO_2 In filter [mg/min]**	**RuO_4/RuO_2**
#1 1227°C 5 l/min air Stainless steel	9.5	0.016	0.40?	0.04?
#2 1227°C 5 l/min air Alumina	8.9	0.437	1.04	0.42
#3 1227°C 5 l/min air Tracer	11.0	0.011	1.31	0.01
#4 1427°C 5 l/min air	25.4	0.055	8.82	0.01
#5 1227°C 10% O_2 + N_2 5 l/min	6.6	0.016	1.69	0.01
#6 1227°C $AgNO_3$ -feed 5 l/min air	10.1	0.579	2.61	0.22

Depending on the experimental conditions 12–35 wt-% of the released ruthenium was transported to the outlet filter as RuO_2 particles. At high release temperature the fraction of ruthenium aerosol was high (35%) due to the higher cooling rate of the gas flow, which increased the RuO_2 particle formation in the gas phase. The fraction of ruthenium aerosol was also high (25%) at low oxygen concentration. Less ruthenium deposited in this experiment probably, because the gas phase concentration of RuO_3 was lower. Feeding silver seed particles (experiment #6) increased the mass flow rate of ruthenium particles transported to the filter by a factor of about 2 as compared to other similar experiment (#3). The number concentration of silver particles at the sampling point was about twice the number concentration of ruthenium particles.

Representative TEM images of the ruthenium dioxide particles collected in gas phase are presented in Figure 85. From the figures we can see that the particles are needle-shaped and crystalline.

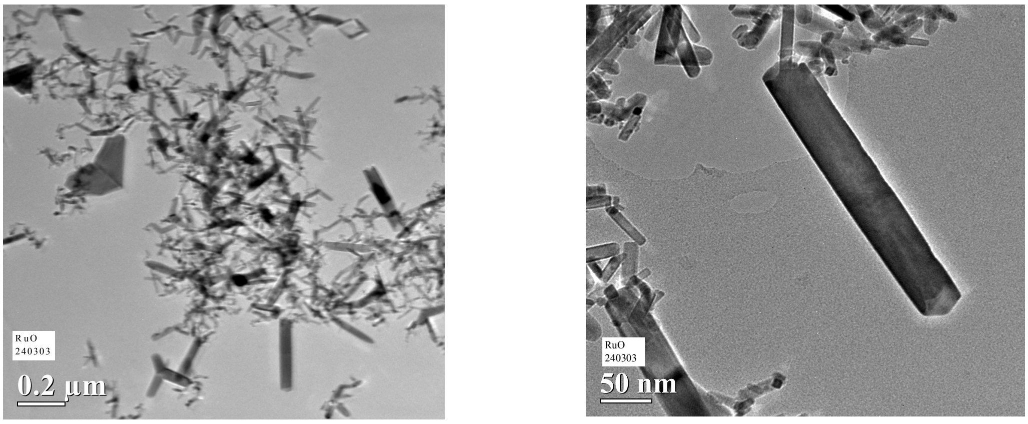

Figure 85. TEM images of RuO_2 particles.

In Figure 86 the deposition profile of ruthenium downstream of the evaporation crucible is shown. This measurement was done in experiment #3 using a radioactive tracer. As the mass balance was closed in this experiment, it was assumed that all ruthenium, which was not found in the trapping bottle, in the filter or in the evaporation crucible in other experiments was also deposited on the tube surfaces. The release crucible was located at -45 – -25 cm and the furnace ended at 0 cm. No deposition can be seen at the location of the release crucible or immediately downstream of it. In the distribution three peaks can be observed. Because RuO_3 exists at neglectable amounts at temperatures below 700°C and because most ruthenium is expected to be released as RuO_3, the deposition in the first peak is most likely caused by thermal dissociation of RuO_3 to RuO_2. The second peak located just downstream of the outlet is caused by thermophoretic deposition of RuO_2 particles. The third peak may be due to decomposition of RuO_4 on the surface of the steel pipe. The gas temperature downstream the furnace is also shown in Figure 86. However, it is not known whether it is the gas or the wall temperature that is more important for the processes taking place in the system.

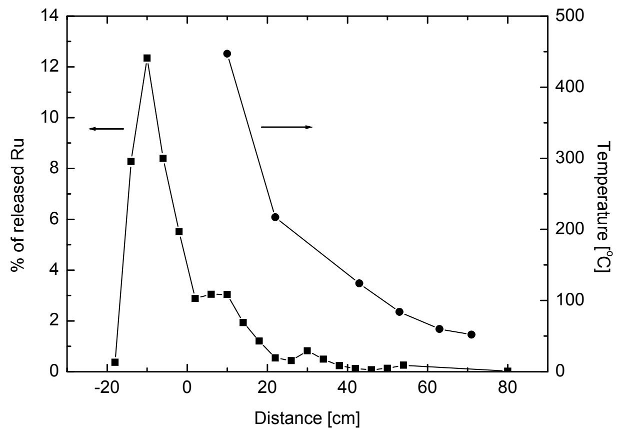

Figure 86. Distribution of Ru downstream of the evaporation crucible. The furnace ends at 0 and the release crucible is located at -25 – -45 cm. The gas temperature is also shown.

In all experiments most of the released ruthenium was deposited on the tube surfaces (65–88%). The deposition mostly took place on the alumina tube within the furnace. Less deposition (65%) at high release temperature was due to the higher cooling rate of the gas flow, which caused the rate RuO_2 particle formation in the gas phase to increase. Due to the difference in diffusion velocity, the particles do not deposit as fast as RuO_3 vapour. Low oxygen concentration also decreased ruthenium deposition (74%), probably because the gas phase concentration of RuO_3 was lower.

Conclusions

In these experiments the transport and speciation of ruthenium at high temperature oxidising conditions was studied. The fraction of gaseous ruthenium reaching the trapping bottle was, in completely dry atmosphere with stainless steel tube, 0.1–0.2% of the released ruthenium. When alumina was applied as tube material or the atmosphere was saturated with water vapour at ambient temperature, the fraction of gaseous ruthenium reaching the trapping bottle increased to 5% of the released. This is close to the release fraction of RuO_4, if the system is assumed to be in thermodynamic equilibrium. This leads to the conclusions that decomposition of RuO_4 is very limited in these two cases and that RuO_2 does not catalyse the decomposition of RuO_4.

A major part of the released ruthenium was deposited on the tube surfaces, most of it inside the furnace, due to thermal decomposition of RuO_3. In the decomposition of gaseous ruthenium species needle-shaped RuO_2 particles were formed. Of the released ruthenium 12–35% passed through the system as RuO_2 particles. Seed particles and a steeper temperature gradient increased the mass flow rate of RuO_2 particles.

However, further experiments would be needed in order to understand this complex matter completely. We are planning to carry out further experiments with different air and steam mixtures. In these experiments the effect of temperature and flow rate on ruthenium transportation and speciation in humid conditions will be studied. Also the effect of steam partial pressure and the effect of seed particles are to be studied in more detail. The catalytic effect of RuO_2 on RuO_4 decomposition, which has been claimed to have taken place in other studies, requires further investigations. Some of the planned experiments are to be carried out using online radioactive tracer techniques.

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14. Development of aerosol models for NPP applications (AMY)

14.1 Aerosol model development for nuclear applications

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Abstract

AMY-project concentrates on understanding and modelling on deposition-resuspension phenomena of aerosols in pipe flow. The aim is to develop a calculation model that could resolve the current deficiencies in the aerosol deposition modelling in turbulent flows, and to implement the models into the tools that are used for calculating the fission product behaviour and release in severe reactor accidents. These tools are APROS SA, which is used for simulating the severe accident phenomena and progression of the accident, and SaTu (support system for radiation experts), which is originally designed to estimate radiation levels and radioactive releases during the accident situation.

In addition to the deposition-resuspension model, other important models are to be implemented in the tools mentioned above. Revaporisation of deposited fission products from primary circuit surfaces may increase the releases into the reactor containment and further into the environment, and thus the phenomenon should be taken into account. To the SaTu system, models for estimating the environmental consequences will be implemented, as well, and the system will be modified to be able to describe nuclear power plants other than the Loviisa plant. Another important feature for source term calculations in PSA level 2 analyses is implementation of the uncertainty calculation environment in SaTu.

Introduction

The most important radionuclides that are released during a severe reactor accident are transported in rector circuit mostly as aerosols. The deposition of these fission products in the pipe systems, especially in containment bypass sequences when the release route goes through narrow pipes, plays an important role in reducing the environmental releases from a nuclear power plant. The deposition in a pipe system is a complex phenomenon including the removal of deposited material either by revaporisation or by resuspension.

The aim of this work is to reduce the uncertainties in aerosol behaviour in pipe systems, and to build a model for resuspension to be used in nuclear reactor applications. The new model, in addition to other related models, will be implemented in codes used for modelling the behaviour of fission products during a course of a severe reactor accident.

AMY project consists of three separate tasks: Bypass sequences, APROS SA and SaTu. In bypass sequences, aerosol deposition in bypass route piping is studied. To complement the experimental data on relevant phenomena some experiments are carried out. Based on experimental data a model for deposition-resuspension in flow condition relevant to the bypass sequences is developed. With APROS SA the aim is to complement the deposition models and validate the fission product models implemented in the code. The SaTu system includes deposition models for auxiliary system pipes, which are revised within the project. A calculation environment is developed to study uncertainties in source term evaluation in PSA 2, and a model to estimate environmental consequences is implemented in the system. In addition, the system will be modified to be applied for a NPP other than Loviisa.

Bypass sequences

The pre-existing knowledge on aerosol resuspension in pipe flows was first gathered in a literature review [1]. This information was used to make a suggestion on the resuspension model and to plan the experimental matrix.

The aerosol deposition is studied in Horizon facility (see Figure 87), which consist of a test section of a horizontal steam generator model originally from the PACTEL facility and additional aerosol generation and measurement devices as well as steam supply equipment, and with PSAero facility (Figure 88) that is a single tube facility with more flexible parameter variation possibilities than Horizon. Both of these two facilities have already been used in the SGTR project within the EURATOM's 5th Framework Programme, but the PSAero facility has been extensively modified to suit better the experiments planned to be carried out in the AMY project.

Two experiments with Horizon were carried out in the AMY project during 2003 [2, 3] to complement the data from earlier experiments. The deposition profiles in single tubes in experiments with high Reynolds numbers (70000–140000) are shown in Figure 89. The deposition profiles are about what was expected, but the total deposited fraction was very small in each of these experiments. The deposition models show almost total loss of the aerosols within the tube, but due to resuspension only small fraction of the material permanently remained in the tube.

The earlier experiments with PSAero showed that at a constant flow velocity the deposited material reached an equilibrium state. Resuspension was enhanced with an increasing flow velocity, and above a certain flow velocity all of the deposits were removed from the tube surfaces (Figure 90). The resuspension model would include information on the deposition history with deposition velocity, maximum flow velocity during the formation of the deposits and some information on the adhesive forces of the deposit layer.

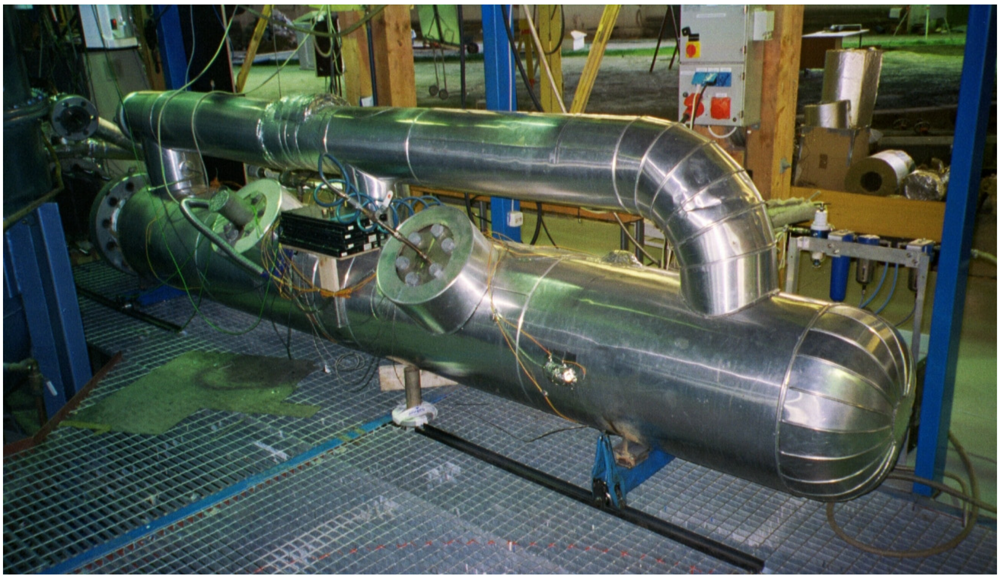

Figure 87. The test section of the Horizon facility (originally one of the steam generators from the PACTEL facility).

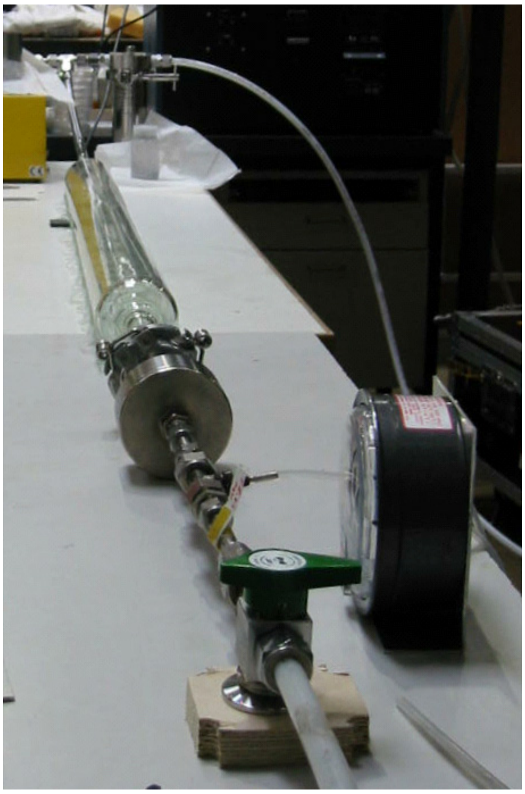

Figure 88. The PSAero facility.

Exp.	DF
05-A	1.02
06-A	1.23
07-A	1.30
08-A	1.11
10-A	1.22
11-A	1.03
12-A	1.02

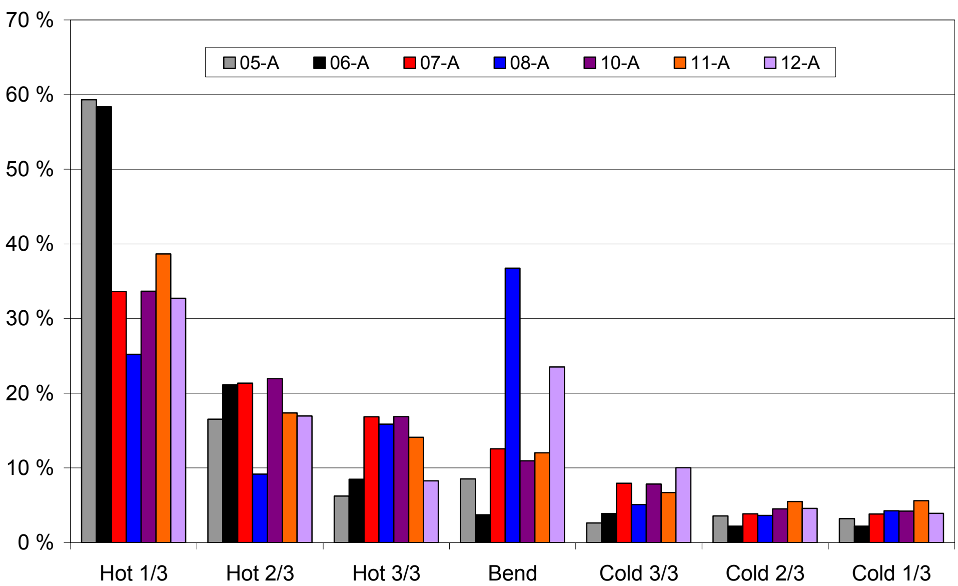

Figure 89. Deposition profiles in Horizon experiments. The figure shows the fraction of deposits in different locations of a tube. The total decontamination factors (DF) are shown on the left.

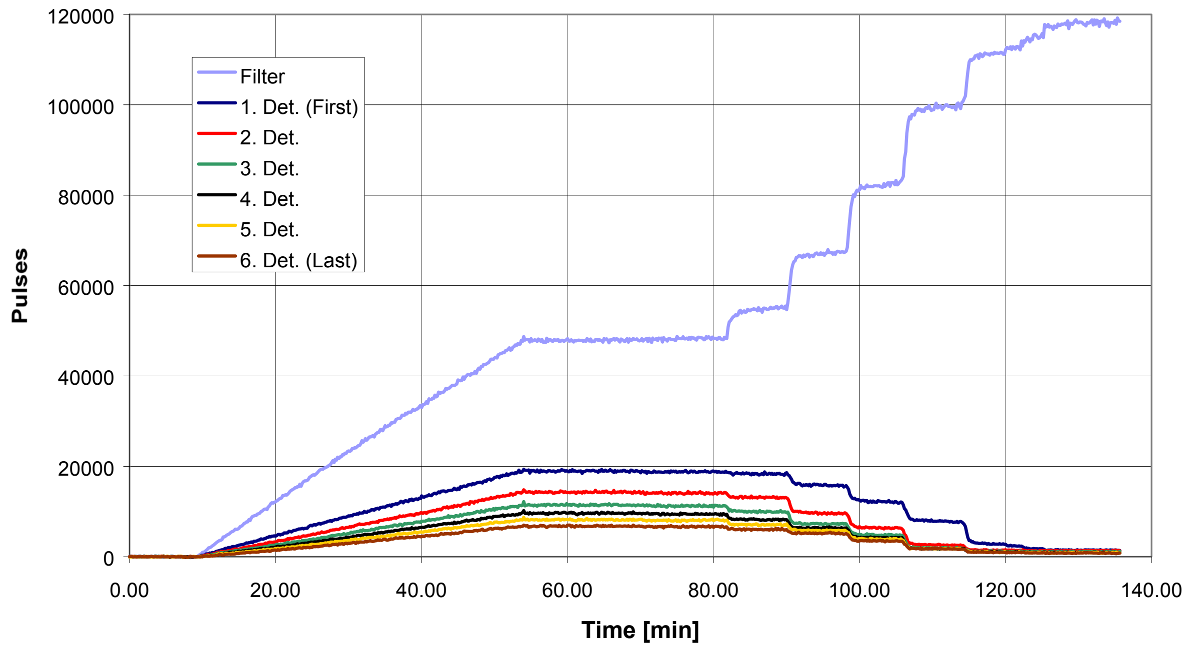

Figure 90. Results from a PSAero experiment with stepwise increase of the flow velocity. The steps show that deposition is removed from the tube surface by resuspension when increasing the flow velocity, but stabilised rapidly to a new equilibrium state.

In addition to the pipe deposition experiments, deposition at the entrance of a perforated plate was studied. The aerosol deposition was considered to be able to form blockages at narrow orifices in the flow path, and thus decrease the flow through pipings that could be considered in bypass sequences resulting in smaller fraction of aerosols to be directed in the bypass route.

To study this phenomenon, CsI and CsOH aerosols were injected through a perforated plate at high flow velocities [4], and the pressure was measured at the outlet side of the plate. The gas was ejected from the outlet by a vacuum pump, and thus the decrease in the pressure level showed clogging of the plate holes. It was observed that addition of CsOH into the aerosol material resulted in formations of looser the deposits and clogging was not as efficient as with pure CsI particles.

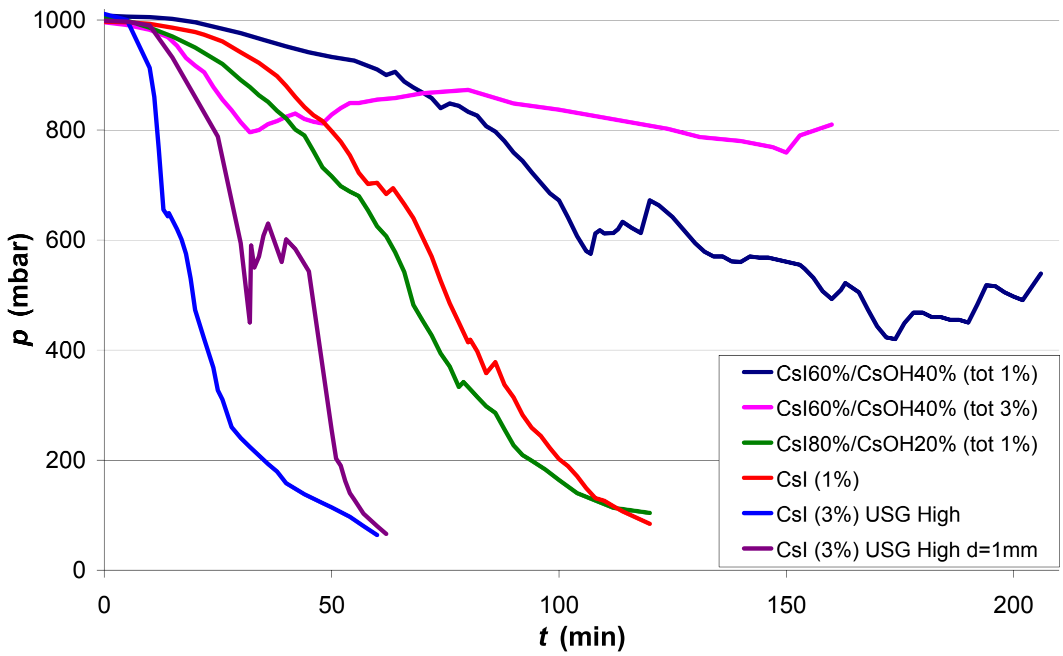

Figure 91. Pressure at the outlet side of the perforated plate as a function of time with the hole diameter of 0.5 mm (1 mm in one exp.). The lower the pressure, the more clogging has been formed. The addition of CsOH into the aerosol material resulted in looser deposits and less clogging was observed.

APROS SA

APROS SA includes fission product transport and deposition models, and the work in AMY project aims at validating the models implemented in the code by separate tasks for primary circuit and containment calculation. Additional models to calculate sedimentation in the primary circuit and the revaporisation from deposits have been implemented in the code, and the work with the resuspension model is still underway.

The aerosol behaviour in the primary circuit has been compared with experimental results from primary circuit model in PHEBUS FPT1 experiment, and the results for iodine are shown in Figure 92. Revaporisation plays an essential role in the hot leg, and all of the iodine is revaporised from the hot leg surfaces, whereas little iodine remained in the experiment. Sedimentation has also some effect on the results, since the flow velocities are rather low. The results from caesium are more or less similar with those from iodine.

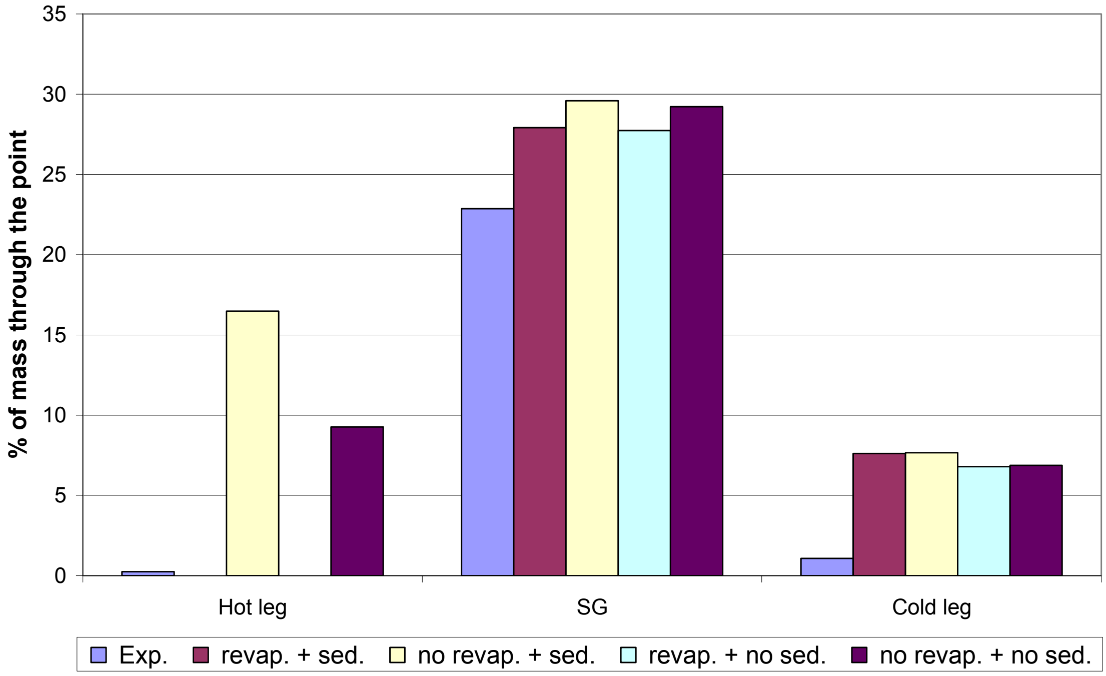

Figure 92. APROS SA calculation results compared with experimental data from PHEBUS FPT1.

The aerosol models in the containment will be validated towards experiments carried out with the VICTORIA facility [5, 6].

SaTu

In order to evaluate radiation levels at the plant and releases of radionuclides during an accident situation at the Loviisa plant a support system SaTu has been developed as a Microsoft Excel application. The system has been modified to support release calculations in the PSA level 2 evaluation. In the AMY project, models for revaporisation and resuspension are to be implemented in SaTu. Furthermore, in order to estimate environmental consequences, a model for radioactive releases in the environment is to be included in the system. In PSA studies, the effect of uncertainties on the results should be estimated, as well. To achieve this requirement, an uncertainty calculation environment has been built for the SaTu system [7, 8]. A view on the input sheet of SaTu is shown in Figure 93 and a sample of results from the uncertainty calculation in Figure 94.

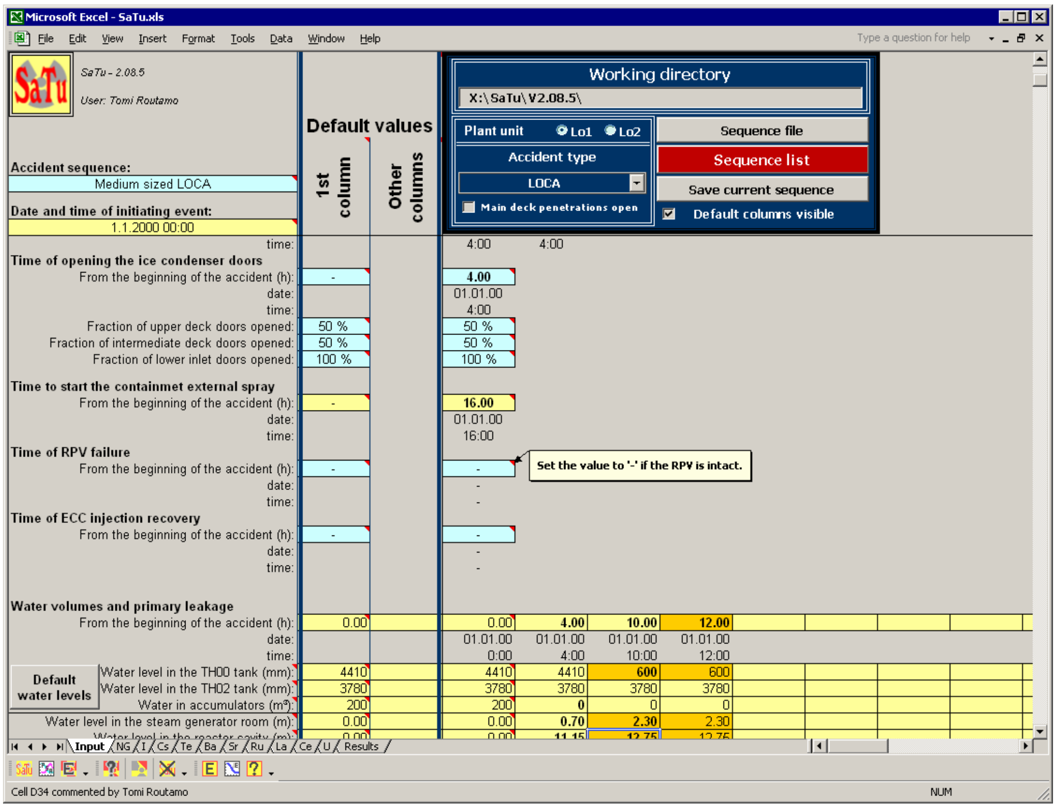

Figure 93. A view from the SaTu input sheet.

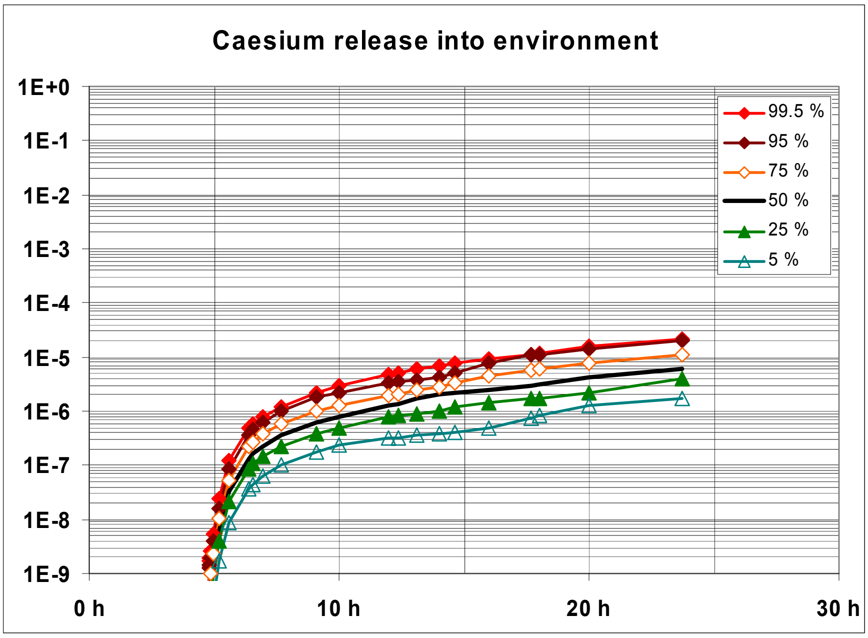

Figure 94. A sample output from the SaTu uncertainty analysis.

In AMY project, the SaTu system will be further modified to be applied in calculations for other nuclear power plant than Loviisa.

Conclusions

As the project is still underway, the main results from the AMY project are yet to come, but it can already be seen that the development of the resuspension model improves the modelling of aerosol removal from the gas in pipe systems. After validation, the new models implemented in systems calculating the fission product behaviour in severe accident conditions result in more realistic evaluation of environmental releases.

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15. Emergency preparedness supporting studies (OTUS)

15.1 OTUS summary report

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VTT Processes

Main objectives

The objective of the OTUS project was to improve emergency preparedness at the Finnish nuclear power plants with two specific tasks. The first is studying radiation levels in rooms and in the vicinity of the power plants and accessibility of them in case of a severe accident during maintenance and refuelling outage when radioactive substances can be transported into different parts of a nuclear power plant and its environment, because the lid of a reactor pressure vessel and different material transfer gates of a reactor containment building may be open. The second task was studying the effect of sea breeze on the dispersion of atmospheric discharges of the power plants during an accident and methods to assess the dispersion during such a specific weather condition.

Main results

In the first task the PSA2 analyses done for the refuelling and maintenance outage of the Olkiluoto nuclear power plant served as a starting point. Primary importance was to get insight of the maximum radiation dose rate levels at the site (Figure 95) and further to assess possible accessibility of working areas. First the accident sequence was selected and the released activities into the power plant and its environment were calculated. Then radiation levels as well as working times could be estimated based on the existing computer codes and other simplified methods. Main interest was to concentrate on the accident sequences, which are essential from a point of radiation doses. Source data was provided by STUK. A report was written comprising the methods used to define accident case, calculations done, results obtained and conclusions drawn on the accessibility of the rooms and the site during maximum prevailing radiation dose rates in the case of a severe reactor accident occurring during refuelling outage [2].

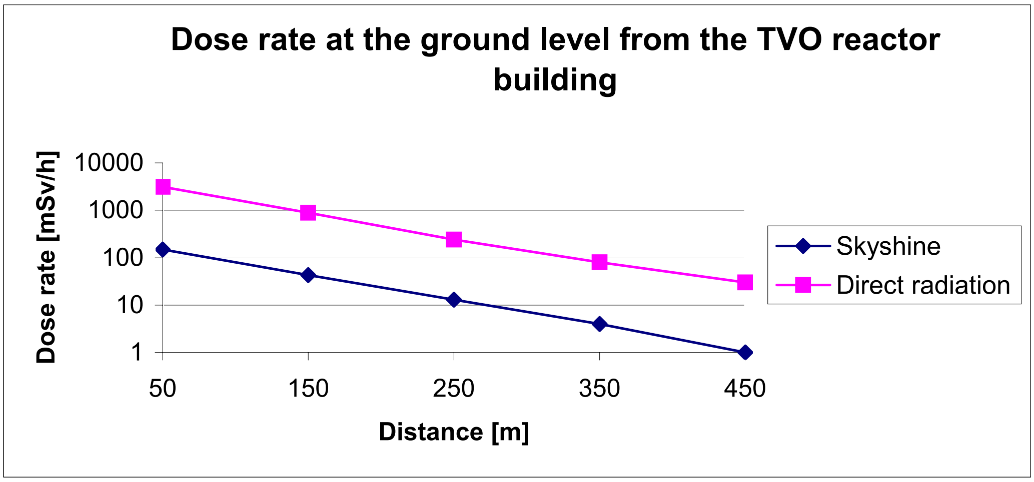

Figure 95. Maximum dose rates outdoors due to a severe accident during service and refuelling outage. Coolant has escaped from the reactor vessel and fuel has melt, consequently discharges are released to the reactor building from open reactor vessel.

In the second task the effect of sea breeze on the dispersion of atmospheric discharges and especially at the Loviisa and Olkiluoto power plants were reviewed based on the published literature. Methods to take this phenomenon into account in emergency response procedures and corresponding dispersion models on the basis of available meteorological data were investigated. The results were summarized in a state-of-the-art report consisting of the concept of the sea breeze as a phenomenon, especially focusing on the Finnish conditions, modelling features and proposals to take the phenomenon into account in the emergency response activities [3].

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16. Archiving experiment data (KOETAR)

16.1 KOETAR summary report

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Nuclear Safety Research Unit

Abstract

In the KOETAR project data and documents of the thermal-hydraulic experiments performed with different facilities at Lappeenranta University of Technology are saved, checked, and archived. Some of the data and documents related to former research programs are still on media that are not compatible anymore with the hardware and software in use today.

The work is done following the priority classification of the experiments updated in the SAFIR research program in 2003. The checked data and documents are archived in the STRESA database of Lappeenranta University of Technology and in CD/DVD disks.

The archived data and documents can be used in nuclear safety research to validate thermal hydraulic and computational fluid dynamics codes. The data is also suitable material for planning and understanding the future experiments and for educational purposes. It can also be used as a Finnish contribution in international co-operation projects. The data and the documents in the STRESA database can be accessed easily through the Internet.

Introduction

At Lappeenranta University of Technology thermal-hydraulic experiments have been performed with different facilities since 1975. Several hundreds of experiments have been carried out. There are about 200 of them from the experiments just with the PACTEL facility.

Data and documents are stored on several media format from CD disks to printed papers. Some of the data and documents are even on media that are not compatible anymore with the hardware and software in use today.

Most of the results from the experiments carried out in the public research programs are owned by VTT. There are also experiments made to customers. The results of those experiments are owned naturally by the customer.

When the experimental activities related to nuclear safety research were transferred from VTT to Lappeenranta University of Technology in the beginning of 2001 the responsibility of managing and archiving the results of the former experiments was assigned for the founded Nuclear Safety Research Unit. However, own funding for the research unit from the university was not available to perform this task. To solve the problem, funding (about 50–60 k€/year) was sought from the public research programs.

The archiving began in the FINNUS research program and continues in the SAFIR research program in 2003–2006. Most of the tools and the methods for the work were created during the FINNUS research program. The results and the tools of the EU CERTA project [1] (2000–2002) are also used in the work.

Main objectives

The objective of the KOETAR project is to save, check, and archive data and documents of the thermal-hydraulic experiments performed with different facilities at Lappeenranta University of Technology.

The work is done following the priority classification of the experiments updated in the SAFIR research program in 2003. The checked data and documents are archived in the STRESA database[5] (http://www.et.lut.fi/yty/stresa) maintained by the Nuclear Safety Research Unit at Lappeenranta University of Technology and in CD/DVD disks. The data and the documents in the STRESA database can be accessed easily through the Internet.

Main results

Data and documents of the thermal-hydraulic experiments performed at Lappeenranta University of Technology have been saved from original media, checked, and archived in the STRESA database of Lappeenranta University of Technology and in CD disks. The CD disks will be replaced later with DVD disks.

In the first stage of the archiving project, the existing PACTEL data saved on DAT tapes in binary format were found, converted into ASCII format, and stored temporally in CD disks to wait further processing. This was done first because the workstation used for the data acquisition was near the end of its lifetime. Data from the experiments carried out with the other facilities are on media (except the paper copies) that are not compatible anymore with the hardware and software in use today. Most of the tools and methods for the archiving work were created next.

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[5] **S**torage of **T**hermal-**Re**actor **S**afety **A**nalysis Data (STRESA) database is developed at JRC Ispra

The archiving has been done following the priority classification updated in the SAFIR research program in 2003. Usability of the data has been checked first. Then the available volt data has been converted to engineering units and all the measured channels have been checked. Inconsistent data found in the check-up have been discarded. The checked data and the documents related to the experiment have been archived then in the STRESA database and in CD disks.

Data from the experiments carried out with the other facilities than PACTEL (REWET facilities, VEERA, and separate effect facilities) have been scanned from the existing paper copies into PDF format. The results of some VEERA experiments were found also in usable digital format in floppy disks. The scanned data curves can be digitized. None of them has been digitized yet because it would require more resources than is available in the KOETAR project.

No major problems have occurred in archiving yet. The time and work needed to convert, check and archive an experiment depends strongly on the amount of measurement channels in use during that particular experiment and on the similarity of the data file structure (number, position and labels of channels in the file) between the experiments in the test series. If there are no major changes from one experiment to another in the file structure, the whole test series can be archived with reasonable work after processing the first experiment of the series. With changes in the number, position and labels of the measurement channels between experiments considerably more work is needed to get the test series archived.

The spectrum of the data formats in the experiments has caused a need to improve the archiving tools every now and then. Also some improvements to the plotting capabilities of the archiving tools have had to be done to make it easier and faster to check hundreds of measured channels per experiment. In future, some problems might be expected especially in finding the existing data and documents of the REWET experiments.

The STRESA database of Lappeenranta University of Technology contains now about 350 different kinds of experiments including the experiments archived outside the KOETAR project. In the KOETAR project about 300 experiments have been archived so far during the years 2002–2004.

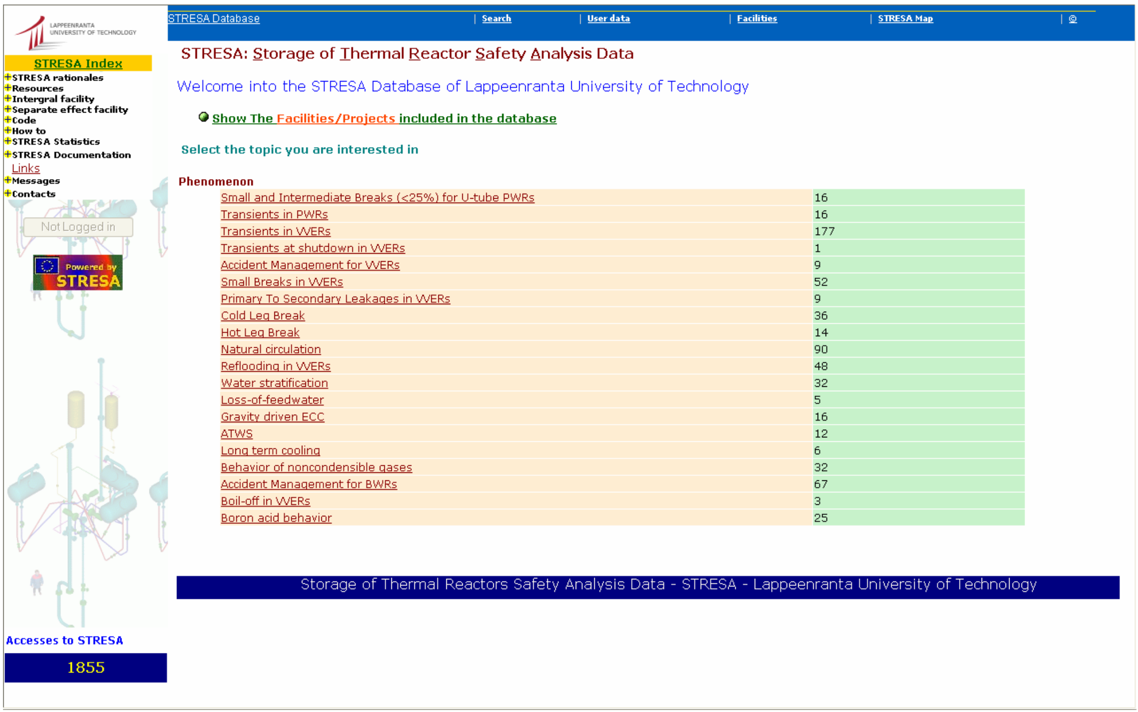

Figure 96. View of the STRESA database.

Applications

The archived data and documents can be used in nuclear safety research for the validation of thermal hydraulic and computational fluid dynamics codes. The data is also suitable material for planning and understanding the future experiments and for educational purposes. It can also be used as a Finnish contribution in international co-operation projects. The data and the documents in the STRESA database can be accessed easily through the Internet after granted permission. Reasonable requests for accessing the material can be sent to the Nuclear Safety Research Unit via the database web pages. The material of the experiments made to order are given only with the permission from the customer who owns the results.

New material from a wide range of research topics is added into the database regularly as experimental work in the Nuclear Safety Research Unit at Lappeenranta University of Technology is still continuing. The archiving tools created in the KOETAR project are thus very useful also in processing and archiving data of future experiments.

Conclusions

At Lappeenranta University of Technology thermal-hydraulic experiments have been performed with different facilities since 1975. Several hundreds of experiments covering a wide range of research topics have been carried out. There are about 200 experiments just with the PACTEL facility. The checked experiment data and documents have been archived in the STRESA database maintained by the Nuclear Safety Research Unit at Lappeenranta University of Technology and in CD disks. The CD disks will be replaced later with DVD disks.

The STRESA database of Lappeenranta University of Technology contains now about 350 different kinds of experiments including the experiments archived outside the KOETAR project. In the KOETAR project about 250 experiments have been archived so far during the years 2002–2004.

The archived data and documents can be used in nuclear safety research in many ways. For example, several thermal hydraulic and computational fluid dynamics codes can be validated against the checked experiment data in the archive. The database contains also good material for planning and understanding the future experiments and for educational purposes. The data can also be used as a Finnish contribution in international co-operation projects.

The data and the documents in the STRESA database can be accessed easily through the Internet after granted permission. Reasonable requests for accessing the material can be sent to the Nuclear Safety Research Unit via the database web pages. New material is added into the database regularly.

The archiving tools created in the KOETAR project are used in the Nuclear Safety Research Unit at Lappeenranta University of Technology also in processing and archiving the data of current and future experiments.

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17. Condensation pool experiments (POOLEX)

17.1 Condensation pool experiments with steam injection

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Abstract

The condensation pool experiments (POOLEX) project has focused on studying phenomena occurring in large water pools when steam/gas mixture is injected into liquid. In a Boiling Water Reactor (BWR) power generation system, the wetwell pool serves as the major heat sink for condensation of steam in case of a primary system blowdown from a pipe break, or in case of a dedicated steam venting of the primary system. Several experiment series with DN80, DN100 and DN200 blowdown pipes have been carried out with a scaled down test facility designed and constructed at Lappeenranta University of Technology (LUT). The initial system pressure of the steam source before the blowdown has ranged from 0.2 MPa to 2.0 MPa and the pool water temperature from 8ºC to 61ºC. During the DN80 and DN100 series the data acquisition system has recorded measurements with a frequency of 20 Hz and during the DN200 series with a frequency of 7782 Hz. Depending on the used blowdown pipe diameter, initial pressure level and pool water sub-cooling, both a stationary steam jet that forms a conical pattern in the pool and large steam bubbles that develop and condense at the pipe outlet have been observed. Furthermore, pressure pulses of the MPa range have been registered inside the blowdown pipe during the DN200 series when the pool water temperature has been close to 10ºC. The experiments have shown that the frequency band of the standard transducers and the frame rate of the standard video cameras are not sufficient to reveal the full nature of the investigated phenomena. Upgrading the measurement system has thus been an absolute prerequisite for the actual detailed tests.

Introduction

The common feature of current BWRs is the use of large condensation pools with a venting system for the mitigation of the immediate consequences of a conceivable Large Break Loss-of-Coolant Accident (LBLOCA) such as a main steam line break. Also, in certain advanced light water reactor concepts, during emergency cooling conditions, mixtures of steam and non-condensable gas are blown into a pool of water via an open pipe. Pressure pulses generated by the collapse of steam bubbles due to rapid condensation, either at the pipe outlet or inside the pipe, may cause considerable loads or even damages when they impact upon a structure.

The phenomena taking place in the condensation pool after an internal pipe rupture in the containment have already been investigated at LUT in the FINNUS/TOKE project, where the behaviour of non-condensable gas (air) bubbles in the condensation pool and their effect on the performance of an ECCS strainer and pump was studied [1]. The SAFIR/POOLEX project focuses on experiments where steam (plus a small fraction of air) is injected into a water pool. The same test rig that has been used in the earlier experiments with air has been slightly modified for the steam experiments.

The main goal of the project is to increase the understanding of the phenomena in the condensation pool during steam injection. These phenomena could be connected to bubble dynamics (bubble growth, upward acceleration, detachment, break-up), to pressure oscillations or to the condensation rate of steam. To achieve this understanding, these phenomena have to be measured with sophisticated instrumentation and/or captured on film with high-speed cameras or. For example, to estimate the loads on the pool structures by condensation pressure oscillations the frequency and amplitude of the oscillations has to be known. Furthermore, strains of the pool wall at exactly defined locations have to be measured accurately for the verification of the structural analysis.

Experiment results of the POOLEX project, supposing that they are exact and of high-quality, can be used for the validation of different numerical methods for simulating steam injection through a blowdown pipe into liquid. Experimental studies on the process of formation, detachment and break-up and the simultaneous condensation of large steam bubbles are still sparse and thus the improvement of models for bubble dynamics is necessary for the reduction of uncertainties in predicting condensation pool behaviour during steam injection [2]. Some of the methods are applicable also outside the BWR scenarios, e.g. for the quench tank operation in the pressurizer vent line of a Pressurized Water Reactor (PWR), for the bubble condenser in a VVER-440/213 reactor system, or in case of a submerged steam generator pipe break [3]. The development work of 3D homogenous two-phase flow models for Computational Fluid Dynamics (CFD) codes can also be assisted by the POOLEX experiments. Furthermore, the coupling of CFD and structural analysis codes in solving fluid-structure interactions could be facilitated with the aid of load measurements of steam injection experiments.

With the aid of high-speed video captures, to be used in the future experiments, the validity of correlations for steam bubble size and break-up heights as a function of total volumetric flow-rate and of pool sub-cooling could be investigated. In determining condensation rates during bubble formation direct measurement of heat and mass transfer is desirable, but virtually impossible. However, the process of direct-contact condensation of large steam bubbles in water is well suited for visual observation. Interfaces are macroscopic and well visible. To some extent, condensation rates can be determined indirectly from volume rates-of-change estimated from video images [2].

Pre- and post-analysis of the loads on the pool structures are necessary for the safe performance of the experiments. Therefore, the experiments have been modelled with CFD codes at VTT Processes. Loads calculated with CFD simulations have then been transferred to structural analysis and the stresses in the pool structures have been evaluated at VTT Industrial Systems. These tasks have been executed in the SAFIR/INTELI project [4].

Condensation pool test rig

The condensation pool test facility is a large cylinder shaped pool with an open top and a conical bottom. The inner diameter of the pool is 2.4 m and the height 5.0 m. Steam needed during the experiments is produced with the steam generators of the nearby PACTEL test facility [5]. Steam is led through a steam line (standard DN80 and DN50 steel pipes) towards the water pool. A sketch of the test rig is presented in Figure 97.

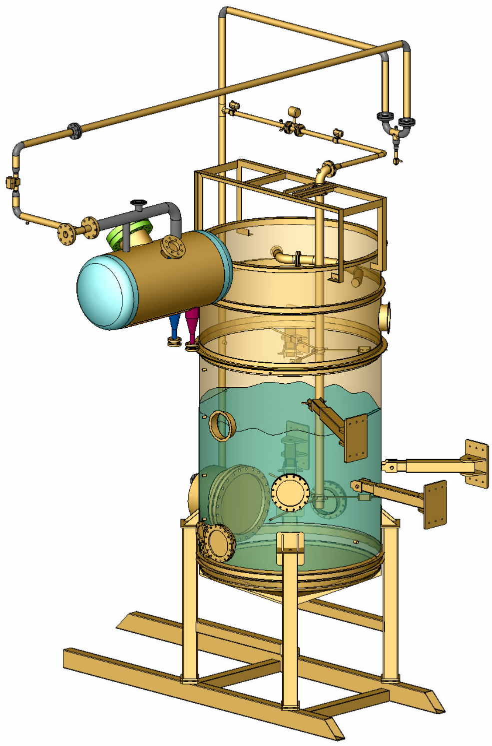

Figure 97. POOLEX test rig for steam blowdown experiments.

DN80, DN100 and DN200 stainless steel pipes have been used as blowdown pipes. These uninsulated pipes are 4.0 m long and their lower ends are 1.2 m above the bottom of the pool. The pipes are placed inside the pool in a non-axisymmetric location, i.e. 300 mm from the pool centre. Table 16 lists the main dimensions of the test rig compared with the corresponding dimensions of the Olkiluoto plant.

Table 16. Test rig vs. Olkiluoto 1 and 2 BWRs.

	Test rig	**Olkiluoto 1 and 2**
Number of the blowdown pipes	1	16
Inner diameter of the blowdown pipe [mm]	214.1*	600
Pool cross-sectional area [m^2]	4.5	287.5
Water level in the pool [m]	3.5	9.5
Pipes submerged [m]	2.0	6.5
A_{pipes}/A_{pool}x100%	0.8	1.6

* used in the DN200 series (Ø 110.3 mm and Ø 84.9 mm pipes have been used in the DN100 and DN80 series)

The test rig is equipped with thermocouples for measuring steam and pool water temperatures and with pressure transducers for observing pressure behaviour in the blowdown pipe, in the steam line and at the pool bottom. Steam flow is measured with a vortex flow meter in the DN50 steam line. Additional instrumentation includes four strain gauges on the pool outer wall. Visual observation of the experiments is done with the help of four digital video cameras through the pool windows. The cameras are standerd 25 frames per second (fps) models.

Experiment programme

The experiment programme in 2003–2004 has consisted of eleven series each including several individual steam blows. The initial system pressure of the steam source before blowdown has ranged from 0.2 MPa to 2.0 MPa and the pool water temperature from 8ºC to 61ºC. During the DN80 and DN100 series the data acquisition system has recorded measurements with a frequency of 20 Hz and during the DN200 series with a frequency of 7782 Hz. Before each experiment the pool has been filled with water to the level of ~3.5 m i.e. the blowdown pipe outlet has been submerged by 2 m. After the remote-controlled shut-off valve in the steam line is opened, the blowdown pipe is filled with steam that immediately pushes its way to the pool. The steam flow may have contained a small amount of air.

Observed phenomena

Depending on the used blowdown pipe diameter, initial pressure level and pool water sub-cooling, either a stationary steam jet in the pool, rapid condensation inside the blowdown pipe or large developing and condensing steam bubbles at the pipe outlet have been observed [6, 7]. With DN80 and DN100 blowdown pipes condensation inside the pipe is very slight at the beginning of the blow before the chugging phenomenon starts to develop. Instead, the steam jet forms a stationary conical pattern below the pipe outlet in the pool and condenses there smoothly. The larger pipe diameter in the DN200 case reduces steam velocity compared to the tests with the same initial parameters (steam source pressure) but on smaller pipes. With lower velocity only a fraction of steam enters the pool and as a result no steam jet is observed below the pipe outlet. Condensation takes place mainly inside the blowdown pipe and therefore steam-water interface moves up and down, see Figure 98. This is true especially when the pool water temperature is low. As the water warms up larger and larger steam bubbles start to form at the blowdown pipe outlet and condense in the pool indicating a shift to the chugging phase. When the pool water bulk temperature is more than 50ºC steam bubbles with a maximum diameter of about two times the DN200 blowdown pipe form at the pipe outlet, see Figure 99. However, each forming bubble condenses rapidly between two frames of the 25 fps video capture.

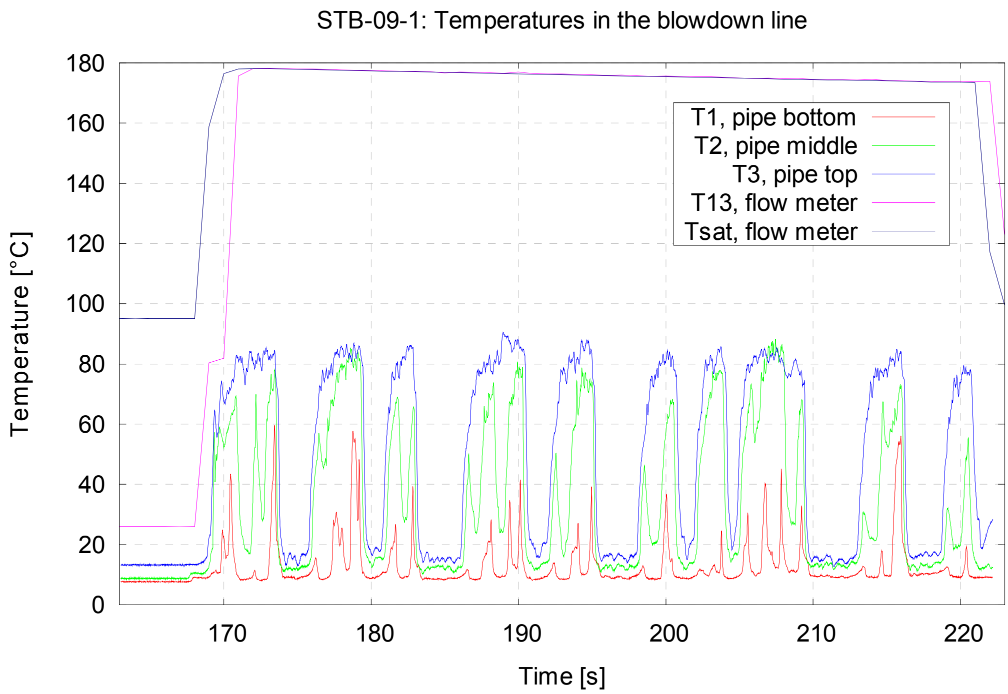

Figure 98. Temperature fluctuations in the blowdown pipe due to the movement of steam-water interface when condensation occurs mainly within the pipe.

Pressure pulses inside the blowdown pipe and at the pool bottom

With the use of a few high frequency pressure sensors and a tuned up data acquisition system in the DN200 series some interesting pressure oscillation behaviour related to rapid steam condensation and water hammer phenomena has been revealed during the experiments. When steam condenses mainly within the blowdown pipe no steam bubbles form at the pipe outlet. Because the condensation process is very rapid, an underpressure develops inside the blowdown pipe. Sometimes the underpressure phase lasts as long as 0.5 seconds. Immediately after the underpressure phase, a condensation induced water hammer is initiated. As a result, pressure transducers have registered very high pulses (maximum 4.8 MPa) inside the blowdown pipe, see Figure 100.

Figure 99. Steam bubble forming at the DN200 blowdown pipe outlet.

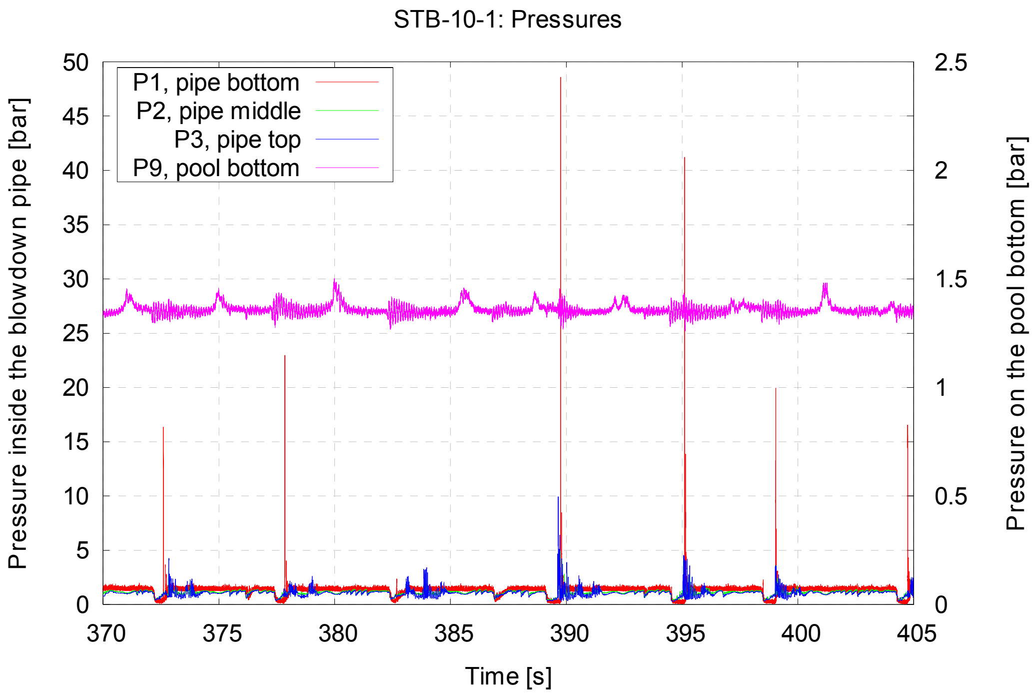

Figure 100. Pressure pulses inside the blowdown pipe due to rapid condensation induced water hammer.

During the steam blows water plugs are pushed from the blowdown pipe to the pool bottom. Water plugs hitting to the bottom and rapid condensation of the steam cause loads to the pool walls. Strain gauges on the pool outer wall, and the pressure transducer on the pool bottom have registered oscillations with the frequency of 10 Hz. The largest measured stress amplitudes by the strain gauges have been 44 MPa. The maximum pressure pulse on the pool bottom has been 140 kPa.

CFD simulations and structural analysis by VTT Processes and VTT Industrial Systems have shown that combined dynamic and static (mass of water) loading of the magnitude experienced in the DN200 series is notable for certain critical locations of the pool wall, particularly for the bottom rounding and welded seams [4].

Conclusions

The steam injection experiments carried out so far in the POOLEX project have aimed at finding out the safe operating limits of the test rig, defining the requirements for instrumentation, data acquisition and visualization and producing measurement data to be used for verification purposes of load estimation and structural analysis in the SAFIR/INTELI project.

The experiments have shown that the frequency band of the transducers and the speed of the standard video cameras are not high enough to reveal the full nature of the investigated phenomena. For example, the pressure curves produced by the slower transducers in the DN80 and DN100 series are lacking essential information. Upgrading the instrumentation has thus been an absolute prerequisite for the actual detailed experiments. A new, fast data acquisition system, a high-speed video camera and high frequency instrumentation has been selected and purchased based on the results of the preliminary experiment series, and on the review conducted about different alternatives. The purchase of the new data acquisition system is expected to greatly enhance the possibilities of the research group to measure accurately the complex, high frequency thermal hydraulic phenomena occurring during steam/gas blowdown and condensation.

The system behaviour is clearly different with the DN200 blowdown pipe than with the DN80 and DN100 pipes. With smaller blowdown pipes condensation inside the pipe is very slight at the beginning of the blow before the chugging phenomenon starts to develop. Instead, the steam jet forms a stationary conical pattern at the pipe outlet and condenses there smoothly. With the DN200 pipe only a fraction of steam enters the pool and condenses there. Condensation takes place mainly inside the blowdown pipe. This is especially true when the pool water temperature is low.

Some interesting pressure oscillation behaviour related to rapid steam condensation and water hammer phenomena has been revealed during the DN200 experiments. When steam condenses mainly within the blowdown pipe an underpressure develops and a condensation induced water hammer is initiated. Pressure transducers have even registered 4.8 MPa pulses inside the blowdown pipe.

CFD simulations and structural analysis by VTT Processes and VTT Industrial Systems have shown that combined dynamic and static loading of the magnitude experienced in the DN200 series is notable for certain critical locations of the pool wall.

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18. PACTEL OECD project planning (PACO)

18.1 PACTEL OECD project planning

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Abstract

OECD launched the SETH project to investigate issues relevant for accident prevention and management and to ensure the existence of integral thermal hydraulic test facilities. The facilities included in the SETH project are PKL from Germany and PANDA from Switzerland. At the early stages of the SETH project an idea was raised to exploit the PACTEL facility in a similar OECD project.

Without any external funding the analytical work in the required extent would not be possible within Lappeenranta University of Technology, the party responsible of operating PACTEL. This fact directed the PACO project proposal to be conducted for the SAFIR programme. The aim of the PACO project is to prepare a project proposal to OECD of a PACTEL related project. To attain this objective some preliminary analyses have to be performed to ensure the relevancy of the proposed topic.

The low power situation, i.e. midloop state was chosen to be the topic in the PACO studies and project planning basis. The plan is to use PACTEL to examine vertical steam generator behaviour during the midloop operation and the following loss of residual heat removal system transient. Such a possibility is acknowledged with special alterations to PACTEL.

The APROS code version 5.04.07 was selected as a tool for the pre-analyses. The "virtual" simulation of the chosen experimental situation would give a preconception on the phenomena to be expected and the progression of the transient.

Originally the PACO project was planned to continue only for a few months, ending up with the project proposal to OECD during the summer time 2004. During the pre-calculation process it became obvious that the time expected was not enough to establish good pre-calculation results. The reasons for this relates to time used to learn and adapt the use of the chosen code, improvements and corrections in modelling as well as the code ability to manage the special conditions defined for the project topic. Another aspect on completing a final proposal was the matter of changing situation on the OECD project application prospects mainly connected to the schedule changes.

These two facts prolonged the PACO project time schedule to be concluded at the end of the year 2004.

The final deliverable of PACO is the proposal of an OECD PACTEL project in which all the knowledge gathered from the analytical work will be utilized. The project proposal will include suggestion of few predefined tests, but also a choice to have one or more tests to be defined by the project participants.

Introduction

OECD launched the SETH project to investigate issues relevant for accident prevention and management through containment and primary circuit tests. The SETH project was introduced to ensure the existence of integral thermal hydraulic test facilities. The facilities included in the SETH project are PKL from Germany and PANDA from Switzerland. At the moment the PKL part of SETH is finished and the PANDA experiments are in their final stage, as scheduled. A continuation for the PKL part of SETH has been introduced as the OECD PKL project and it is now under its way.

At the early stages of the SETH project an idea was raised to exploit also PACTEL in a similar OECD project. First supposition of a SETH project successor was that it would utilize several thermal hydraulic facilities under the same project. Later it became evident that instead of one project, an approach with several separate projects is more probable. To apply such an OECD project utilizing PACTEL has to be well and carefully planned. This means that the topic of research has to be widely interesting and the experimental results are estimated to contribute important information on safety issues. To emphasize the substance of the project proposal, careful pre-analyses of the planned experiments and phenomena expected to discover have to be carried out to support the proposal.

Without any external funding the analytical work in the required extent would not be possible within Lappeenranta University of Technology, the party responsible of use of PACTEL. This reality directed the PACO project proposal to be conducted for the SAFIR programme. The PACO project is established to prepare a project proposal to OECD of a PACTEL related experimental and analytical project. The PACO includes preliminary thermal-hydraulic analyses to ensure the relevancy of the proposed topic.

The low power situation of midloop state was chosen to be the topic in the PACO studies and project planning basis. The captivating plan is to use PACTEL to map out possibilities to examine vertical steam generator behaviour during the cold shutdown situation, the midloop operation, followed by loss of the residual heat removal system (LRHRS). Such a possibility has been acknowledged with alterations to PACTEL.

The computational pre-analysis, the thermal-hydraulic simulations, are considered to hold one important role in the experimental planning of the PACO project. The APROS code version 5.04.07 was selected as the computational tool for the PACO pre-analyses. APROS was selected for the analyses, even if the implementation of the two other possible codes, CATHARE and RELAP, was estimated to be easier.

Selected transient for pre-calculations

The main objective of the PACO project is to produce a project proposal to OECD of the PACTEL related experiments. The planned scenario includes midloop state, presence of non-condensable gases and the following LRHRS transient. A significant element is the new experiment assembly planned, e.g. use of vertical steam generators. To attain the objective pre-analyses are needed to ensure the relevancy of the topic.

Originally PACTEL has been planned to simulate the primary side of the VVER-440 type Loviisa Power Plant [1]. With PACTEL, numerous experiments have been carried out over the years. In the first PACTEL assembly three long narrow steam generators were used in horizontal position [2]. Later on, new steam generators replaced them, again in horizontal position. It is possible to introduce vertical steam generators in PACTEL, Figure 101. This is enabled using the first old steam generators in vertical position. Two of the present PACTEL loops are planned to be used, whereas one of the loops remains isolated from the rest of the system. This kind of assembly allows widening the field of research to the application areas connected to western type of PWRs.

The objective of the preliminary pre-analyses in the PACO project is to get an insight and preconception on phenomena occurring during the LRHRS transient following the midloop state. At the midloop state the water level in the primary system is reduced to the level of hot leg connection. This means draining of primary water and replacement of that with non-condensable gas, i.e. nitrogen in this case. The conditions in the primary system are in normal atmospheric pressure and low temperatures. The status of midloop is a premiss for the power plant outage enabling the maintenance actions in reactor system. During the midloop situation the primary water level is thoroughly at low levels and the sections over that level are filled with non-condensable gases.

The midloop state in PACTEL means that the water level is at about 8.75 metres above the bottom of the lower plenum. This state means also the filling of the upper plenum, the steam generator U-tubes and pressurizer with non-condensable gas. The transient starts as the core begins to heat up, because of the LRHRS transient. The following phenomena are main subject of research. Special attention is paid to analyzing the flow behaviour inside the steam generator U-tubes, behaviour of non-condensable gases as well as possible start-up and nature of the natural circulation and system behaviour.

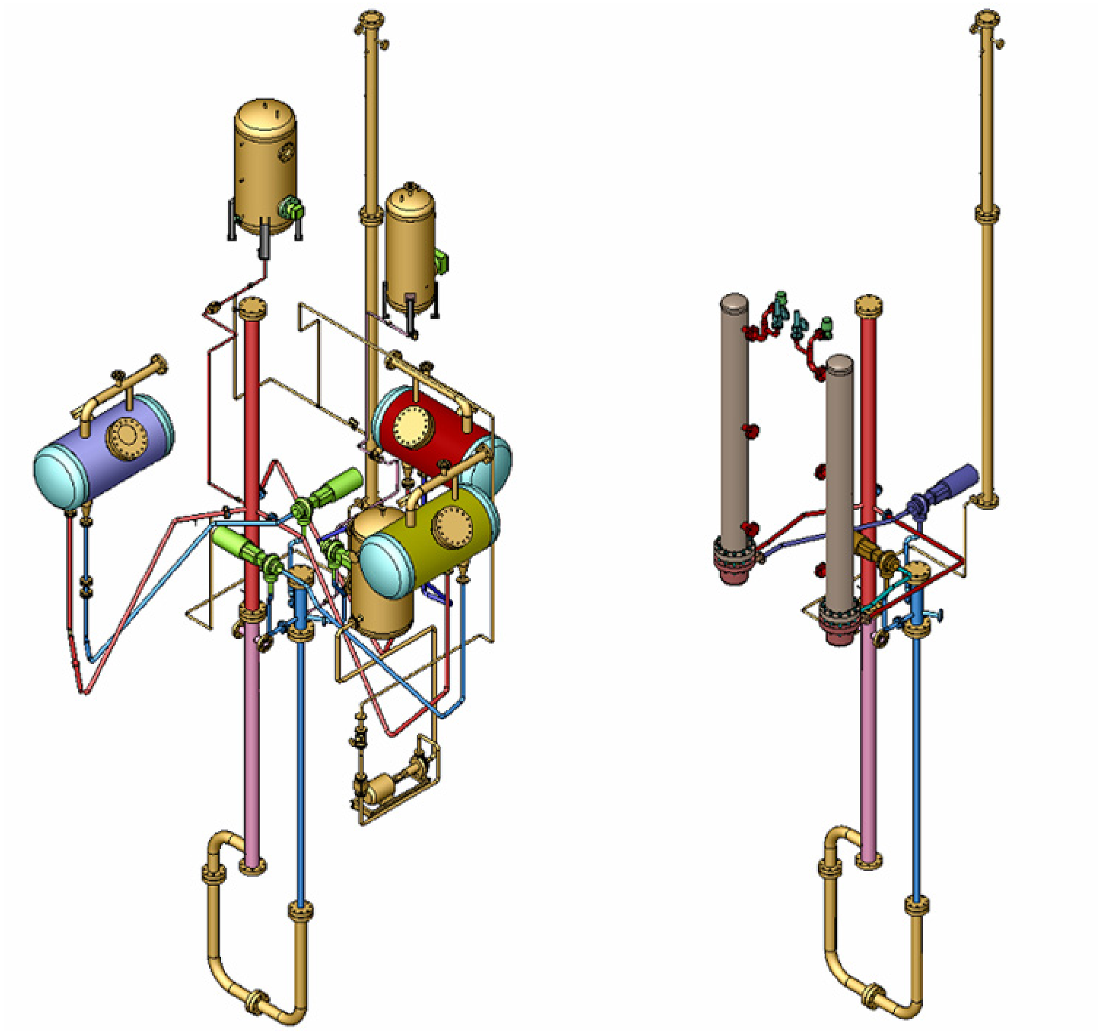

Figure 101. Present PACTEL facility and planned new PACTEL set-up.

Originally the PACO project was planned to continue only for a few months, ending up with the project proposal to OECD during the summer time 2004. During the pre-calculation process it became obvious that the time expected was not enough to establish good pre-calculation results. The reasons for this delay relates to time used to learn and adapt the use of the chosen simulation environment, improvements and mere corrections in modelling as well as the code ability to manage the special conditions defined for the project topic. Another aspect on completing a final proposal was the matter of changing situation on the OECD project application prospects mainly connected to the schedule changes. These two facts prolonged the PACO project time schedule to be concluded at the end of the year 2004.

Now, the main task is to perform final simulations to examine the phenomena occurring and possible parameters regulating the system behaviour. This should give an idea on actual project and experimental planning, whether the topic is of great interest and feasible, and what to expect during the actual experimental scenario. Only after this the primary objective, i.e. project proposal planning, can be realised.

Pre-calculations

The PACTEL pre-calculations have introduced models for the new planned set-up of PACTEL. An old PACTEL simulation model has been modified to present the new two-loop construction with the vertical steam generators. There are two models, simplified (Figure 102) and more detailed one. The difference between the two is in the description of U-tube construction, i.e. the simplified model uses more lumped modelling scheme. The detailed model can give a more detailed insight of possible discrepancy between different length steam generator tubes, regarding behaviour of coolant and non-condensable gases, condensation and vaporisation phenomena.

One feature of the PACO project, that became evident during the pre-calculation process, was the strong need for further code development. This was not set to be one of the primary objectives for this project. Nevertheless, the code development and further modifications on the code version used became a crucial aspect to ensure progression of the pre-analysis. The cold shutdown situation, e.g. the midloop situation means the presence of non-condensable gases in low temperature and pressure environment. It had been seen already in earlier other projects, that the code capabilities should be improved. Such situations are cases involving the non-condensable gases, low pressure situations and rapid and drastic changes in temperatures. The modifications to the APROS code version 5.04.07 used in the PACO project were entirely performed by VTT code developers. In regard to the PACO pre-calculations the concurrent code development work resulted to delivery of the special modified version of APROS. Utilisation of this modified version led to a success, i.e. to the non-diverging simulation calculations in the PACO project, in late October 2004.

Today, using the recently modified version of the APROS 5.04.07 code, the simulations with the simplified PACTEL model gives general results, which partly correspond to earlier calculations. Anyhow, some differences can be seen, partly because of the new code version as well as some the improvements in the simulation model, e.g. modified pressurizer conditions adjustment during the midloop state.

Before the actual transient, the secondary side of the loop 3 is drained from water. The temperatures over the whole facility are about 60°C and the pressure is equivalent to the normal atmospheric pressure. The upper volumes are filled with nitrogen gas. The transient starts as the core power is set on corresponding to the LRHRS situation. The core part starts to heat up. The primary system stays closed during the whole transient.

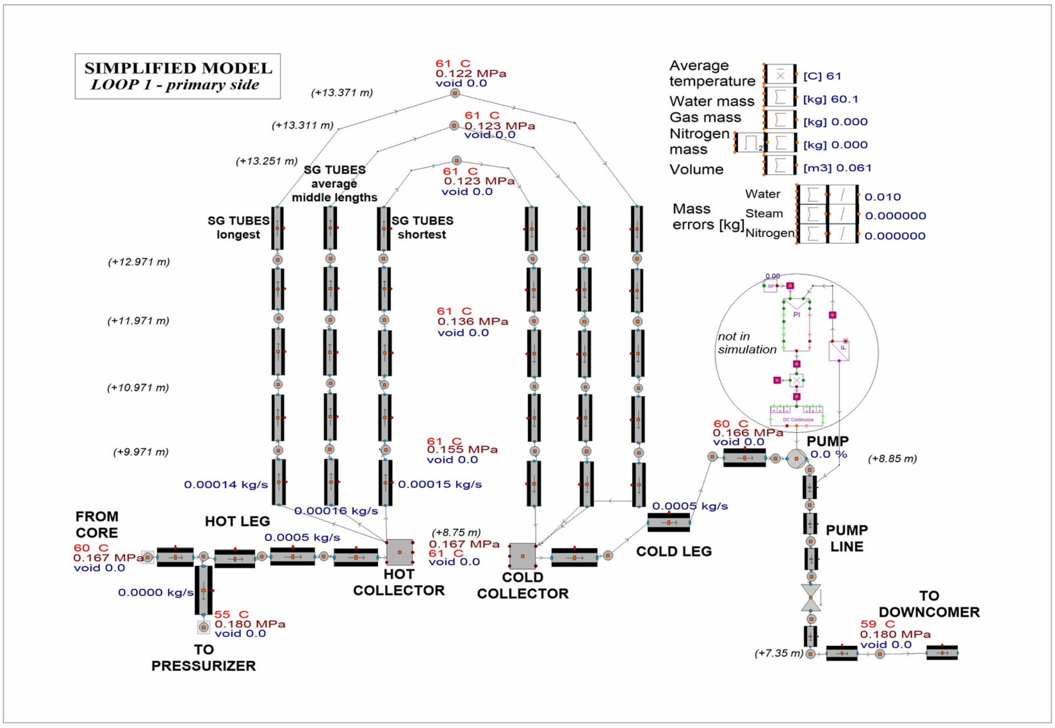

Figure 102. PACTEL simulation model: loop 1 primary side.

Generally, the transient progresses as follows. The heat up of the core leads finally to the vaporization of the water in the core part. The produced hot steam pushes the nitrogen in the upper plenum to flow to the loops as well as to the pressurizer. The pressurizer water level rises. The loop 3 with empty secondary side can not correspond to the gradually warming steam that flows into the primary U-tubes. Finally all the U-tubes of this loop are blown through. The hot steam flows through the cold leg, pushing the nitrogen via downcomer to the loop 1.

Simultaneously the other loop 1, with the secondary side almost full of water, starts to take over the heat transfer. Calculations show that this steam generator can manage the whole heat transfer, hence finally fully corresponding to the heat up of the core. Anyhow, the pressure rise before this special natural circulation state can be several bars, and the core structure maximum temperatures rise even to about 200°C, before the steady natural circulation. Observations on the loop 1 indicate the filling of U-tube with condensated water. Cold water in downcomer lower parts as well as the lower plenum is not significantly taking part to the cooling of the system.

Calculations show the amount of the nitrogen to be one major factor in the heat transfer progress. More calculations have to be performed to establish the sensitivity of this as well as the system behaviour connected to factors such as used core power, water levels in the secondary sides and U-tube structure changes. This work is still going on.

Applications

The developed simulation models can be further improved and used in post-calculation analysis, as the planned new PACTEL set-up experiments will be carried out. With further improvements the models and simulations can be used as an assisting tool in more exact experimental planning, i.e. on planning the text matrix, to map the parameters of great importance as well as to test constructional changes.

The modifications to the APROS code version used in the PACO pre-calculation project have vastly improved simulation calculation performance. This information on achievement of the smooth progressive simulation feature of APROS code implies improved situation of the code capabilities to solve the transient situations of non-condensable gases present. This knowledge should be beneficial when using the code in other similar applications, e.g. actual power plant transients.

The code capabilities can be tested with pre-and post-calculations, against the experimental results to be performed by PACTEL. This aspect gives more insight on the code performance with simulations of corresponding possible situations at real power plant environment. So, code validation could benefit from the planned experiments, connected to low power situations involving non-condensable gases.

The realisation of the proposed project will generate new important safety related experimental data, widen the application area, as well as ensure the use of PACTEL in the future. The OECD PACTEL project will also assist in preserving and enhancing the know-how, and introduce material for pre- and postgraduate studies.

Conclusions

OECD launched the SETH project to investigate issues relevant for accident prevention and management as well as to ensure the existence of integral thermal hydraulic test facilities. The facilities included in the SETH project are PKL and PANDA. The PKL part is finished and the PANDA experiments are soon completed. The OECD PKL project has been introduced to continue the PKL part of SETH. At the early stages of the SETH project an idea was raised to exploit also PACTEL in a similar OECD project.

The PACO project was launched within SAFIR to prepare a proposal to OECD on PACTEL related research program. The chosen experimental scenario includes midloop situation, presence of non-condensable gases and the following transient situation of loss of residual heat removal system (LRHRS). An important factor is the new experimental set-up, e.g. use of vertical steam generators. The computational pre-analysis, the thermal-hydraulic simulations, are considered to hold one important role in the experimental planning within the PACO project, i.e. to get a preconception on phenomena to be expected during the LRHRS transient. The APROS code version 5.04.07 was selected as the computational tool for the pre-analyses.

The pre-calculations have introduced models for the new planned set-up of PACTEL. One feature on the PACO project, that became evident during the pre-calculation process, was the strong need of further code development. This was not set to be one of the primary objectives for this project. Nevertheless, the code development and further modifications on the code version used became a crucial aspect to ensure progression of the pre-analysis. Utilisation of the modified version led to a success, i.e. to the non-diverging simulation calculations in the PACO project in late October 2004. The modifications to the APROS code version used in the pre-calculations have vastly improved simulation calculation performance. This information on achievement of the smooth progressive simulation feature of APROS code implies improved situation of the code capabilities to solve the transient situations of non-condensable gases present.

The final deliverable of PACO is the proposal of an OECD PACTEL project in which all the knowledge gathered from the analytical work will be utilized. The project proposal will include suggestion of few predefined tests, but also a choice to have one or more tests to be defined by the project participants. However, the proposal will be submitted to OECD when the probability of the acceptance of the project is higher than at the moment, e.g. within couple of years.

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19. Interaction approach to development of control rooms (IDEC)

19.1 Development of indicators for integrated system validation

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Abstract

Integrated evaluation of complex artefacts such as NPP control rooms are claimed to require improved validation approaches. A notion of system usability is introduced. It denotes connecting decisions concerning the appropriateness of artefacts to criteria that indicate the system's ability to fulfil the objectives of the activity system and to promote their appreciation in users' situated actions. System usability is evaluated via an operation-oriented evaluation concept. It includes evaluation of user performance and experienced appropriateness of the system. New types of performance evaluation criteria are introduced. The new indicators and criteria may be derived with the aid of a Core-Task Analysis modeling process.

Introduction

In this paper we present our on-going research project that studies the modernisation of the information and communication systems of the two Finnish nuclear power plants. Our particular research questions concern the modernisation of the human-system interface (HSI) in the main control rooms of the plants. The objective of the research is to develop a method that can be used in the integrated system validation (ISV) [1] of the emerging control room solutions. The meaning of integrated system validation is to ensure that the newly developed control room system functions in the intended way to provide the desired level of safety efficiency and user wellbeing of nuclear power production. Thus it is an essential step in the licensing of new and modernised NPP control rooms [2].

The need for the development of ISV methods rises from the current situation in which the validation methods described for example in NUREG guidelines [1, 3] lack some important aspects that are prerequisites for meaningful validation process. The first problem is the *selection of relevant operational situations* for the validation. This problem has been addressed in recent work by the present authors [2, 4]. The second problem is the so-called *indicator* problem. While having determined appropriate indicators or measures for validation a third problem remains. It is the *criterion* problem [2, 4]. How should the criteria of accepting or rejecting control room designs be formed? The meaning of the criteria is to make explicit what kind of performance is acceptable and what is not. The fourth problem is the so called *effort* problem [2, 5]. What is the amount of testing and evaluation that is needed to validate a system? The "graded approach" in validation refers to the levelling of the scope of the test and evaluation activities. The primary factors that influence the grade include: safety significance, operational significance and personnel safety hazard [6].

System usability – Promoting the core-task orientation of user practices

An artefact, i.e. the control system of an NPP, is fit for its use when it promotes meaningful activity in the usage organisation. The usage of a complex cognitive system i.e. the control system of an NPP can be comprehended as an activity system [7] as in Figure 103. The model depicted in figure was developed by Engeström on the basis of the cultural-historical theory of activity [8]. The object/outcome of the activity, and the new possibilities for action that it offers, provide the motivation for the subjects to take actions and define the structure of actions. The artefacts have an important role in determining how the users' situational goals build up. In the activity system the artefacts mediate the relationship between the subjects and their object. Further mediators within the activity system are rules, norms and values, and the division of labour in the organisation. The meaningfulness of the activity may be comprehended by analysing the content of the activity system.

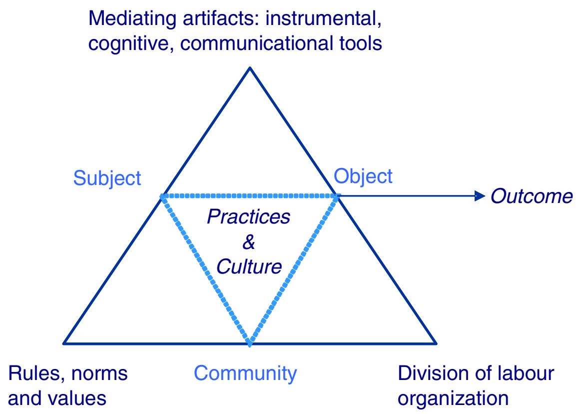

Figure 103. A model of an activity system, modified from [7].

The concept of Core Task. In order to elaborate further what meaningful *activity of an organisation*, and meaningful *actions in an organisation* are, we use the concept of Core Task [9–11]. By core-task we denote the core content of work, which is determined by the object of activity and its aimed results. They constitute the societal meaning of the activity. In peoples' situated actions for reaching the result the intrinsic constraints of the work domain [12] must be taken into account. *Practices* and *culture* emerge as a result of the continuous interplay between the environment and the personnel to reach the targeted, but continuously developing objectives of the work.

The process of defining the core task is called the Core Task Analysis (CTA) [9, 11]. Not only experts of the domain or the human factors specialists but also the operators and other members of the plant personnel have an active role in the core-task modeling process. CTA begins with defining the object and the societal motive of work. The object and societal motive of nuclear power production may, in short, be defined as producing maximum amount of electricity from the nuclear fission under the constraints of balancing safety, efficiency, maintainability and quality [11]. Also the pressures that tend to change the object or the intended outcome should be analysed.

The concept of system usability. We have proposed the concept of system usability as the generic qualification of the appropriateness of complex systems [13]. We exploit the above model of the activity system (Figure 103) and represent the control room system as an information and communication technology (ICT) -based mediating artefact that serves the instrumental, cognitive, and communicational functions [14] of NPP operations. Together with rules, norms, and values, and the organisational structures characterising the power plant operations this tool determines the realisation of actual working practices and culture. Therefore, the criteria of *system usability* of an artefact are manifested in the performance of the whole activity system and its outcome.

System usability should, furthermore, reflect the intrinsic constraints [12] of this operational domain. Thus, for example, in the nuclear domain it is required that the control room system is able to portray appropriately the critical safety, efficiency and personnel health-related constraints of the production. Furthermore, the artefact should promote users' possibilities to take these constraints into account and form meaningful actions, and thus develop *appropriate* work practices. *A system with high system usability induces good working practices on the users*. It also facilitates the development of community knowledge and joint practices.

In the evaluation of system usability in a modernisation context we face the problem that the outcomes of modernisation induce *changes in the whole system.* Hence, adequacy of process control or operator performance cannot simply be considered as independent variables, with regard to which a comparison between the old and the new technology could be made. Instead, it is necessary to develop a way to comprehend the *development trajectory* from the functioning of the old HSI to a balanced and appropriate functioning of the new HSI [15].

Two aspects of evaluation of system usability

Performance-based evaluation. According to the above reasoning, system usability of an artefact must be evaluated by examining the *user practices* it induces. We claim that system usability becomes manifest in promoting the *core-task orientedness* of users' actions. Thus, it is necessary to derive human performance indicators and evaluation criteria that express this core-task orientedness of the operator crews' performance. An indicator development requires a combined process of work domain and core-task modeling, and analysis of operator real performance. The theoretical justification of this approach is described in recent book by Norros [11]. During this conceptual analysis the generic domain and task demands are identified and portrayed in situation specific constraints and behaviours of the operator crews [4]. The performance-based indicators are defined so that they express *the appropriateness of the design solution* by demonstrating how well the *information presentation refers to operationally relevant information*.

Acceptance-based evaluation. Because system usability is conceived as a balanced and appropriate functioning of the whole human-technology system, we hypothesise, further, that the level of achieving system usability also manifests in user acceptance of the artefacts. Thus, we agree with authors that advocate the importance of user acceptance in integrated system validation [16]. We see, however, that acceptance is an expression of the functionality of the whole system, not a separate attribute of the subjects as the above-cited authors conceive it [17]. For this reason the term *"experienced appropriateness"* is introduced to substitute for the "user acceptance".

By including this evaluation aspect of system usability we extend the way of comprehending the roles of the human actor and technology. According to the traditional engineering idea, technology is seen to reduce human failures. Therefore, human action should be replaced by automation. In some situations this is true but a further point of view is also plausible: Automation and human actors may be seen in a compensatory relationship to each other; each have their strengths and weaknesses which should be balanced in good design [18]. This principle holds in many cases but does not explain the phenomenon exclusively. A third, human-environment system approach claims further, that development of technology enables new ways to distribute intelligence in human-technology systems, with the aim to improve human capabilities and possibilities of life. This perspective satisfies the needs for human-centred design and promotes innovative and sustainable technology [see also 19].

The framework for operation-oriented system usability evaluation

A framework for system usability evaluation has been constructed in the IDEC project. The core of the framework is the evaluation model depicted in Figure 104.

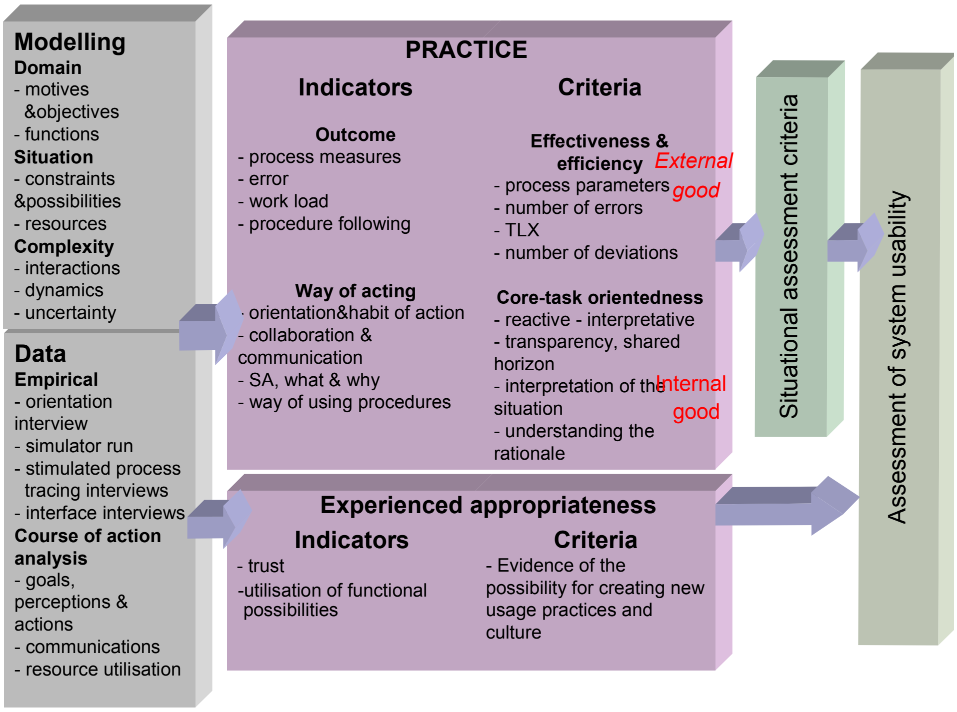

Figure 104. The operation-based evaluation of system usability of complex systems.

The notion of *operation-based evaluation* is used in the framework to indicate the combination of a performance-based and experience-based evaluation of the appropriateness of artefacts. These two evaluation aspects should allow conclusions regarding the system usability of the artefact. Bases for the system usability evaluation are created by, first, modeling the domain and the core task, and, second, by empirical analysis of operators' performance and conceptions (see Figure 104). Different types of modeling and data acquisition methods are used. These methods will be described in detail in the forthcoming report of the framework [20].

The user practices are evaluated from two points of view [4, 15] (Figure 104). An *external* point of view denotes to the scrutiny of the actual courses of action. The users' sequential intentional interactions with the system and the outcome of action a, i.e. process performance, are considered. The courses of action of the operators, which we study with the aid of an array of data collection and analysis methods [15], should first be evaluated with regard to their *successfulness to maintain the process in aimed result critical boundaries*. We use contextually relevant situation-specific output criteria that are defined carefully according to the logic of the core-task modeling. In our studies, we have considered the adequacy of the identification of the loss of process stability, the use of stabilisation methods and the identification of the cause of disturbance [15, 21]. As Figure 104 indicates, we also include errors, work load and following procedures as outcome indicators of performance. It has been claimed, however, that outcome-centred indicators may be insensitive because experienced operators are able to compensate possible problems in the artefacts [16, 22]. Therefore, it should be necessary to analyse how, or in which ways the operators use the artefacts.

Hence a second, *the internal* point of view to practice becomes relevant. It aims at identifying the different ways or modes of using artefacts. It should reveal the habitual and cultural features that the operators' actions express. Drawing on combined behavioural and interview data we assess how the users interpret and take into account the objectives and constraints of the work domain in the situated actions. Thus, we analyse what *interaction demands* the functional constraints put on the human actors, and what *reasons* the actors express verbally or in behaviour. This provides information of the *internal good of practices, i.e.* what is considered appropriate and valued as professional within the community of practitioners [23]. Thereby the *core-task orientedness* of the practices. This is the first aspect of the internal good of practices.

By analysing the human-process interactions, human-human interactions, and the actor's self-control, we acquire a further possibility to value the internal good of practices. We define ways of acting, i.e. habits, that according to the pragmatist theory enable tuning with the environment by establishing a continuous cycling between the states of doubt and belief [11, 24]. Hence, the second criterion for internal good of practices is the *adaptability* of practices that improves the possibilities of the actor to dwell in the environment. A list of 51 criteria of good process control practices were established and used in evaluating operator practices for training and validation purposes [15, 21]. The criteria are behavioural descriptors of ways of decision making, ways of collaborating and ways of coping with problem situations. The descriptors are classified into six categories that express to what extent the operators have in action a) taken into account the global state of the process; b) taken into account the particular nature of the disturbed state of the process; c) promoted a shared interpretation of the situation; d) promoted the unity of collaboration; e) promoted reorientation in problem situation; f) promoted critical evaluation of own resources. Behavioural descriptors are defined for evaluations. The situational evaluation aspect is depicted in the evaluation framework of Figure 104 where the phase of deriving situational assessment criteria is indicated before the final assessment.

The definition of the indicators of experienced appropriateness is an on-going work of the project. It is approached both via users' conceptions and their behaviour. Experienced appropriateness would in the former case be evident in some experiential qualities, like trust in the artefacts or ease of use. Behavioural indicators would inform of the breath and depth of use of the functionality of the artefacts.

Before being able to make assessments of the control rooms it is necessary to define *acceptance criteria* for all above mentioned indicators. The definition of criteria is an on-going task of the project and it will deserve further attention in the forthcoming work of the project.

Conclusions

There are several novel features in the IDEC methodology. The first deals with the usage of a functional analysis of the work domain in the definition and selection of test situations. The aim is to improve the representativenes of evaluation results. The second new feature is the use of the functional work domain analysis in a core-task –based definition of performance indicators and evaluation criteria. Improving the outcome-centred performance evaluation with the evaluation of the internal good of practices provides us with information of the generic principles of behaviour, which regulate the operators' actual actions. By this new feature we aim both at higher sensitivity of evaluation and better predictability. The inclusion of user acceptance into the evaluation tool is a further new feature of integrated system evaluation. In our method user acceptance is not tackled via detached inquiry of operators' opinions about the artefacts. Rather, we aim to develop acceptance to an integrative measure of how well the artefact may be embedded in a new way of working. The new practice should combine the possibilities of the artefacts with sufficient user competencies to meet the production objectives of the future work. The described indicators are used to evaluate system usability of artefacts.

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20. Application possibilities of systematic requirements management in the improvement of nuclear safety in Finland (APSREM)

20.1 APSREM Summary Report

Veli Taskinen
RAMSE Consulting Oy

Main objectives

The main objective of the work was to map state-of-the-art Requirements Management (ReM) practices, and to study and present a summary of the practices and application possibilities in the selected application areas (Figure 105). The most important application areas were the nuclear authority (STUK) activities and the modifications and procurement of nuclear power plants.

Specific goals

- to study the Requirements Management research situation
- to study the state-of-the-art practices in Requirements Management of the selected application areas
- to study the application areas of requirements management of STUK and utilities
- to draft the Requirements Management process for FIN5 nuclear power plant case
- to identify Requirements Management standards applicable in the nuclear industry

Main results

- The study on research in ReM indicated that ReM has been conducted mainly in software production, especially in the requirements definition phase of ReM. Short descriptions of the methods developed in the research projects have presented in the APSREM project report. [2]
- It was found that the focus of the identified ReM practices was on the definition of stakeholder requirements and on software production. Essential approaches and the most important practices concerning the ReM for authority use have been identified in the project, and their main principles have been described.

- Safety control of new and operating nuclear power plants was pointed out as application area of ReM practices for authorities and procurement of new nuclear power plants and modification of operating plants for utilities.
- A tentative ReM process has been developed for the nuclear safety control of the FIN5 nuclear power plant.
- About 30 nuclear safety standards have been analysed from the requirements management point of view. The results showed that ReM has been included in different ways and to a varying extent in the standards, and only part of them are good enough for systematic ReM.

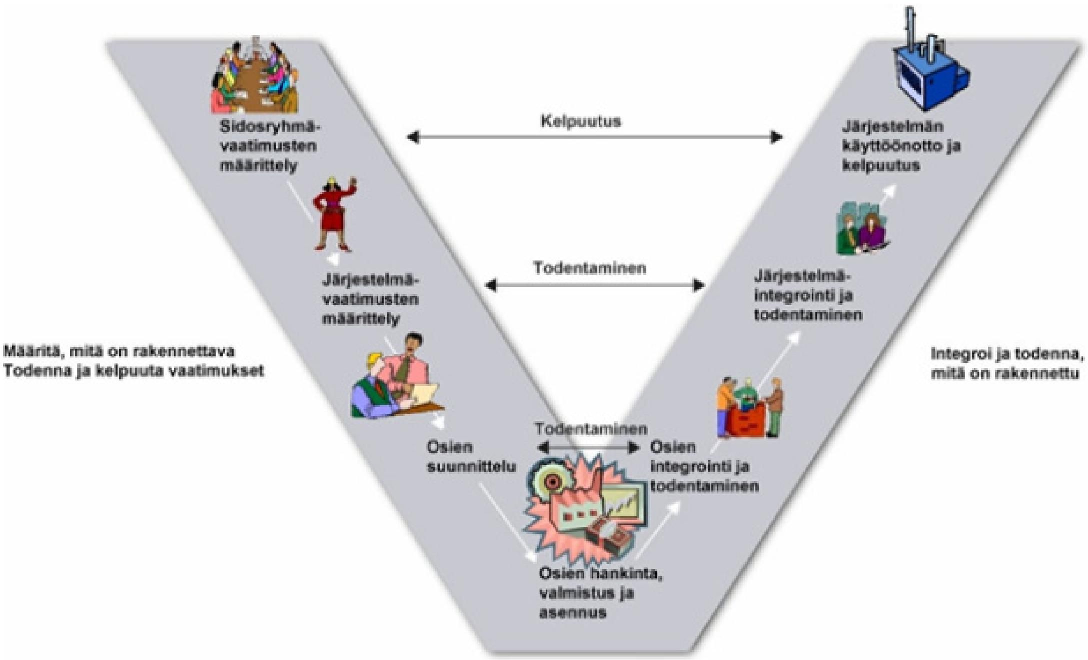

Figure 105. The APSREM project aimed at producing nuclear specific requirements management material in Finnish.

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21. Influence of RoHS-directive to reliability of electronics – preproject (ROVEL)

21.1 ROVEL summary report

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Introduction

The target of this prestudy was to study the necessary changes in electronics due to the so called RoHS -directive (Directive 2002/95/EC of the European Parliament and of the Council, of 27 January 2003, on the restriction of the use of certain hazardous substances in electrical and electronic equipment [1]).

The directive will ban the use of lead, quicksilver, cadmium, hexavalent chromium and flame retardant materials PBB (polybrominated biphenyls) and PBDE (polybrominated diphenyl ethers) in new electronic products put to market after 1 July 2006. The needed changes in materials and processes may alter the product so much that its properties do not correspond present.

Main objectives

Modifications due to EU RoHS -directive

The changes due to the directive were studied by collecting information from Finnish and international research projects. This was done by participating to the following international conferences, the Scandinavian NoNE (No Lead in Nordic Electronics) project final conference in June in Copenhagen and IPC/JEDEC 2004 International Conference on Lead-Free Electronic Assemblies and Components in October in Frankfurt.

Participation in IEC TC45 standardisation work

The development of the new IEC-standard, IEC 62342: Nuclear power plants – Management of aging of nuclear power plant instrumentation and control and associated equipment (45A/441/NP), has continued. The work was done in the IEC TC45 working group WG A10, Instrumentation systems.

Main results

Modifications due to EU RoHS -directive

The so called RoHS-directive is going to prohibit the use of some elements in the electronic industry: lead, chromium 6+, mercury, cadmium, polybrominated biphenyls (PBB) and polybrominated biphenyl ethers (PBDE). The ban of lead is at the moment considered to be most critical to the reliability of electronics.

Lead has been used in the soldering materials because it makes the soldering operation easier. The properties of the lead containing solder joints are also better. Connection technologies of electronic components have a big influence to the reliability of electronic components and also to the end products. That is why each new material and technology requires to be studied before they can be adopted. Higher melting point of lead free materials is a very important parameter. This means a higher soldering temperature and some differences in the production. The components and the printed circuits boards (PCBs) have to tolerate higher temperatures. Also the chemicals that are used in the process have to be designed for the lead free process. The producers of soldering materials offer many alternative lead free solder alloys. The main metal in these lead free alloys is tin. The most common supplements to tin are copper and silver (the SAC-alloy).

According to the published research results a reliable solder joint can be made using lead free materials and processes. There are still some possible reliability problems. Mostly these problems are due to some unsuitable materials and chemicals that are still on the market. And some problems come purely because the process window of the lead free process is tighter than in the commonly used process at the moment. This makes the process control more demanding which can cause reliability problems.

Participation in IEC TC45 standardisation work

The development of the new IEC-standard, IEC 62342 Nuclear power plants – Management of aging of nuclear power plant instrumentation and control and associated equipment, was finished in to a committee draft [2]. The next phase will include development of detailed standards for selected components (e.g. connectors and cables).

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22. Organisational culture and management of change (CULMA)

22.1 CulMa summary report

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VTT Industrial Systems

Abstract

The aim of the CulMa (Organisational culture and management of change) -project (2003–2006) is to enhance the understanding of the effects of organisational culture, organisational changes and the organising of work on the safety of nuclear power production. The goal is to develop methods and models for taking organisational culture into account in a manner that all the criteria for an effective organisation (safety, productivity and health) are fulfilled. Contextual Assessment of Organisational Culture (CAOC) methodology has been developed to promote this. CAOC aims at assessing the elements of the organisation guiding the daily practices and decision making, and identifying development targets. Case studies in Finnish and Swedish NPPs have focused on three themes: forms of organising work, organisational assessment, and management of change. The power plants have to tackle with these issues in their daily practices. This has been demanding for numerous reasons including the lack of behavioral science expertise at the companies. CulMa project has aimed at providing the NPPs with sound scientific knowledge of work and organisational behavior, e.g. the relation of organisational climate on human performance or the challenge of coping with the demanding aspects of work. Especially the maintenance function has previously received little attention regarding the psychological issues.

Introduction

Organisational culture refers to the practices, principles and meanings that guide the daily work and decision making in the organisation. The culture has an influence on the overall safety and reliability of the plant operation. Culture guides e.g. investment decisions, content of training, formulation of procedures and everyday working practices on the shopfloor level. The elements of culture are often so self-evident or subconscious that they are not questioned until something critical happens. The subconscious nature of culture also makes change complicated. New technologies and practices may face resistance or be utilised in a manner that was not the original purpose. For this reason, the impacts of changes in some part or function of the organisation on organisational culture and safety have to be taken into account. These impacts have manifested clearly in the accidents, where the central reason has not been an individual human error but rather the practices, norms and culture of the organisation, e.g. in the case of Challenger explosion [11] or the Piper Alpha offshore platform fire [12].

NPPs in Finland are facing pressures for organisational changes due to e.g. deregulation of the electricity markets and the change of generation (both technology and people). The building of the fifth reactor unit in Finland as well as the forthcoming licensing of Loviisa NPP prompts the power companies to restructure their practices and processes. Utilisation of new information systems, training facilities, and management philosophies could in principle allow a more efficient and motivating organising of work. At the same time the companies should be able to ensure and prove their reliability and safety to the general public and the regulator. The power plants have to tackle with issues related to the above challenges in their daily practices. The challenge of managing change has been experienced as demanding at the power plants for numerous reasons including the lack of behavioral science expertise at the power companies. CulMa project has aimed at providing the NPPs with sound scientific knowledge of work and organisational behaviour. Especially the maintenance function has previously received little attention regarding the psychological issues.

Main objectives

The aim of the CulMa-project (2003–2006) is to enhance the understanding of the effects of organisational culture, organisational changes and the organising of work on the safety of nuclear power production. The goal is to develop methods and models for taking organisational culture into account in a manner that all of the criteria for an effective organisation (safety, productivity and health) are adequately considered. The CulMa-project is organised into three themes (see Figure 106).

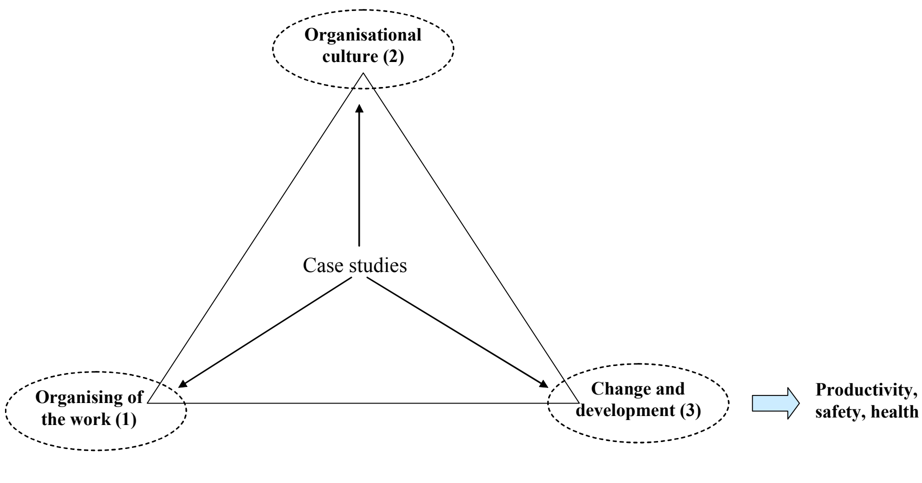

Figure 106. The three themes of the CulMa project.

CulMa-project has continued the work started in FINNUS/WOPS (1999–2002) [6] project aiming at developing a methodology for assessing the organisational and safety culture at NPPs. The methodology is called Contextual Assessment of Organisational Culture (CAOC). During CulMa CAOC will be evolved from a scientific research method into an easy-to-apply practical tool for the power companies and the regulator.

CulMa-project seeks to *construct a psychological theory on reliability-seeking organisations and their special features* with the use of the concept of organisational culture. We aim to clarify the psychological questions that the organisational culture has to provide an answer to, including e.g. the following: How does a hierarchical organisational structure affect the sense of personal responsibility? How to promote a realistic sense of control among the workers concerning their tasks?

Main results

Organising for safety and efficiency in the nuclear power industry (theme 1)

In the year 2003 the purpose was to produce an overview of outsourcing and the use of contractors in the nuclear power industry. The study was based on a literature survey and a case study carried out at Olkiluoto power plant. The results suggest that in the nuclear power industry external contractors are particularly used for periodic and labour-intensive tasks and that an increasing portion of subcontracted work is carried out within the partnership framework. It was also established that the limited availability of quality products and services in certain industry-specific competence areas, such as reactor, turbine and automation systems, is currently one of the most important constraints relative to the use of contractors. The results were reported in a form of a public research report [1] and a conference paper [2].

In the year 2004 the purpose was to focus on organisational change and development projects implemented at Finnish NPPs and to identify their main drivers, objectives, content, methods of implementation, and results. A case study at Loviisa NPP was conducted. The case study concentrated on the 2002 organisational change. The most important drivers of the change initiative were the ongoing generation change and the perceived need to enhance the effectiveness of plant life management. A decision was made to shift to a new four unit model. Functions were transferred between the different units with a view to break down technology-based divisions, to build more efficient operational entities and to provide the most experienced engineers with an opportunity to focus on plant life management-related issues. In addition, more responsible jobs and vacancies had been arranged for younger employees. The major challenges related to the division of work between the Technology and Maintenance units, and matters relating to supporting the transfer of expertise.

Assessment of Organisational Culture – Methodological Development (theme 2)

The subproject continues the work done in FINNUS/WOPS-project to develop a methodology for contextual assessment of organisational culture (CAOC). In CulMa the methodology is validated and improved. The purpose of the CAOC methodology is to assess the elements of the organisation that guide the daily practices and decision making. The assessment provides information on the development targets and challenges of the organisation. CAOC utilises two concepts, organisational culture and organisational core task (OCT). The OCT concept frames the shared constraints and requirements that all the workers have to take into account in all their tasks. The OCT can be used in assessing the central dimensions of the organisational culture. [9]

The rationale of using the core task modelling in cultural assessment has been published in [4] and the results of the pilot version of the CULTURE-questionnaire in [8]. Another scientific article has been published about the assessment of maintenance culture at TVO and FKA [10]. Articles are used in a dissertation work concentrating on the development of CAOC-methodology to be completed in due time.

Development of working practices and management of change (theme 3)

The subproject aims at developing a framework for utilising the knowledge of the organisational culture in change management and development practices. The aim is to identify and model the critical factors affecting organisational changes at NPPs. Purpose is to enlarge CAOC-methodology towards a more development-oriented approach.

CAOC-methodology has been applied in studying and developing the maintenance work at Loviisa, Olkiluoto and Forsmark NPPs [4, 7, 10]. The aim of the case studies has been to assess how the maintenance culture supported perceiving and fulfilling the demands of the maintenance core task. As a concrete outcome of the study conducted at Olkiluoto a longitudinal research focusing on training and socialisation of newcomers at TVO maintenance was started. Also a measure of the conceptual understanding of the maintenance work and the plant technical specifications was created for evaluating the effectiveness of the training process.

The effect of organisational changes on culture has emerged as an important research question. During CulMa a model for describing the organisational aspects having a potential safety impact in the change process will be created. An important question is how the safety critical organisations differ from other organisations. Are the general work psychological theories applicable to NPPs? We have thus modelled the special characteristic of safety critical organisations [5].

Applications

Due to its contextual and participative nature, the CAOC-methodology acts as an intervention into the culture of the organization. The aim of the research is, therefore, not only to assess the culture, but also to give the personnel new concepts and new tools for reflecting on their organization. Core task modeling prompts the personnel to discuss and make explicit the aspects taken for granted in their daily work. The approach strives to enhance the organisational capability to assess the current working practices and the meanings attached to them and compare these to the actual demands of the organisational core task and in that way change unadaptive practices.

Some specific applications that have been developed in the CulMa-project include the organisational culture survey CULTURE [8], orientation interview method, model of the demands of the maintenance core task [4, 10] and a pilot version of a self-assessment instrument of the maintenance culture and core task.

Conclusions

CAOC-methodology has been applied in case studies and reported in international scientific journals [4, 8, 10]. The concepts of organisational culture and organisational core task appear to work well in describing and assessing complex work such as maintenance of a NPP. CAOC provides important background information for development initiatives at NPPs. Future challenge in terms of development of CAOC methodology is to provide tools for implementing and assessing organisational changes.

No systematic method for assessing the effects of organisational changes exists. Problems common to all changes irrespective of the industry also manifest in NPPs. These include increased uncertainty, unclear responsibilities and an initial decline in performance after the change. The safety critical nature of NPPs also sets unique demands for the organisational change such as a need for organisational redundancies, inherent complexity and the need to prove the safety implications of the change.

Understanding the dynamics of change and the mechanisms of change management in NPPs and the question of the role of personal responsibility are issues needing attention in the future. The change of worker generation is a major organisational change facing the NPPs in the coming years. The challenge has been acknowledged for a number of years and the power companies have prepared for the change in various ways. CulMa-project emphasises the psychological challenges of the change of generation identified in the case studies. The challenges include how to facilitate the sense of personal responsibility and realistic sense of control among the newcomers and how to maintain the motivation of the older employees [3, 10].

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22.2 Organisational challenges of maintenance work at NPPs

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VTT Industrial Systems

Abstract

The proper working of the machinery is critical to NPP safety and productivity. Because maintenance routines and plant modifications are the activities that intervene most with the machinery, they are also the dominant sources of faults. Most of the human factor studies have relied on this fact. Due to the diversity, the temporal and spatial separation of the maintenance tasks, and the numerous competence requirements, focusing on a single task, special situation or a single psychological problem can only partially explain the requirements of maintenance work and the organisational challenges of effective maintenance. We have applied a cultural approach to maintenance work. Our aim has been to model the maintenance task and its psychological requirements and to characterise the features of organisational cultures in three NPP maintenance units. Results imply similarities and differences in the cultures and in the emphasis on the maintenance task. Maintenance activities have been under various restructuring initiatives. These changes have been experienced as stressful among the personnel. The effect of changes on the reliability of maintenance should be considered. A challenge for maintenance is to be able to build organisational structures and practices that would facilitate the fulfilment of the psychological work characteristics.

Introduction

The competence in maintenance consists of different technical fields and requires strategic understanding as well as practical handicraft skills. For example in annual outages the maintenance organizations have to schedule and plan hundreds of work packages requiring multiple technical disciplines. In addition to that, all work has to be coordinated with the operations and be performed according to organizational procedures.

Since maintenance and modifications are the activities that intervene most with the machinery they are also the dominant sources of faults [cf. 4, 6, 14]. Most human factor studies have relied on this fact. They have aimed at classifying, predicting and preventing human errors or minimizing their consequences [5, 6, 12, 13, 14]. Various safety and error management programs have been initiated [4, 8, 14]. The aim to *control and reduce the variability of human performance* is common to most human error approaches. The main solution to performance variability is to standardize, instruct and teach the work tasks. Furthermore, in NPPs the guiding design principles of redundancy (multiple backups for the same function) and diversity (variety in the design principle of the safety functions) are applied to manage error. Work permit procedures, audits and quality assurance are considered as defenses against system disturbances [14]. Reason and Hobbs [14] list the following factors that increase the frequency of maintenance errors: documentation problems (quality of content and presentation), time pressure, poor housekeeping and tool control, inadequate coordination and communication, fatigue, inadequate knowledge and experience, and problems with procedures (out-of-date, non-understandable).

An error-oriented approach to work has its limitations. Rochlin [21] argues that safety means more than the absence of accidents or errors. The meaning of "safety" is also socially constructed and interpreted in the organisation [21]. The above mentioned error-promoting "factors" can be considered as consequences of the cultural conceptions concerning safety and reliability. Further, due to the diversity, the temporal and spatial separation of the tasks, and the numerous competence requirements, focusing on a single task (e.g. electric installation), special situation (e.g. outage) or a single psychological problem (e.g. memory overload, error of omission) can only partially explain the requirements of maintenance work and the organisational challenges of effective maintenance.

We have applied a cultural approach to maintenance work. Our aim has been to model the maintenance task and its psychological requirements and to characterise the features of organisational culture in various maintenance units. This article summarises results from the three case studies in Nordic NPPs from the point of view of the organisational challenges of the maintenance work.

Organisational culture and maintenance culture

Organisational culture refers to the practices, principles and meanings that guide the daily work and decision making in the organisation. The culture has an influence on the overall safety and reliability of the plant operation. The elements of culture are often so self-evident or subconscious that they are not questioned until something critical happens. Weick [23] has described the continual and collective reality-building process constantly taking place in the organization. In this process the meanings of various events are deliberated and a common view is formed based on perpetually incomplete information. Cultural approach thus emphasizes collective issues (and those issues that should be shared) over e.g. individual decision making. Individuals act and make decisions in a social context. The effect of this context can be so strong that the individual is not even aware of making a decision – choosing between alternative ways of acting. [1, 18, 22, 23]

Our methodology (CAOC) utilises two concepts, organisational culture and organisational core task (OCT). The OCT concept frames the shared constraints and requirements that all the workers have to take into account in all their tasks. The OCT concept can be used in assessing the central dimensions of the organisational culture. The organisational practices, values and conceptions are evaluated against what the organisation is trying to accomplish and what demands it has to fulfil in order to survive. The aim of conceptualising the OCT is not to prescribe the structures or practices needed to accomplish the organisational core task. Instead, the aim is to explicate the demands that the organisation has to manage in its everyday activities. The demands can be fulfilled organisationally in various ways. [18]

When studying organizational culture attention should be given also to the dysfunctional solutions, ambiquities, and discrepancies in the organisation, as well as the attempts to solve or cover these [7, 9]. The homogeneity of the culture (widely shared conceptions and assumptions) as such is not always a criterion for good culture. The demands of the OCT dictate whether certain cultural features (differences in opinion, emphasis on self-confidence) are good, bad or insignificant for the effectiveness of the organisation [18]. For example, different opinions can facilitate discussion and be adaptive in fulfilling the demands of safety and reliability [20].

Methods

The Contextual Assessment of Organizational Culture (CAOC) methodology has been utilized in case studies of maintenance culture at Olkiluoto, Loviisa and Forsmark NPPs during 2001–2004 [9, 15, 16, 20]. The theoretical OCT model was created and used in evaluating the characteristics of the organizational culture (Figure 107). We aimed at identifying the strengths and weaknesses of the case organization's culture in relation to its core task. The focus of the assessment was not on explaining causal relations to objective measures (e.g. occupational accidents or number of human errors). Instead, we strove to anticipate the consequences of the current practices, conceptions and assumptions in the given organizations to their ability and willingness to fulfill the OCT [18].

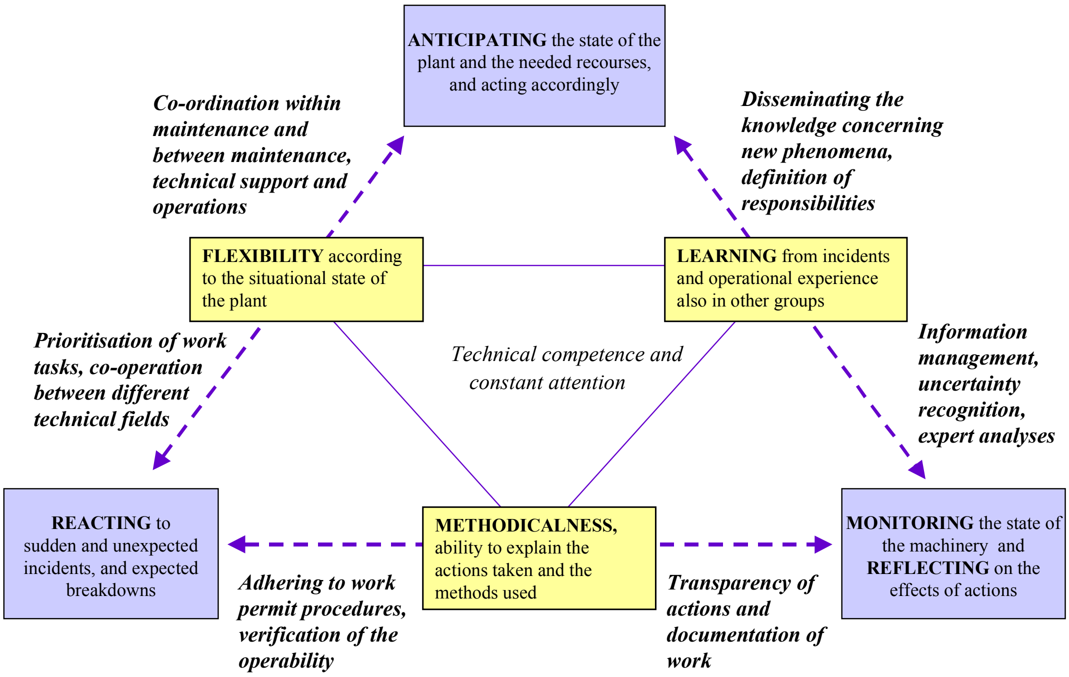

Figure 107. The model of the demands of the maintenance core task, from [9] and [20].

Figure 107 shows the three critical demands of maintenance: *anticipating*, *reacting*, and *monitoring and reflecting*. The organisation has to balance in parallel between these demands. *Anticipating* refers to an intention to predict the state of the plant and the effects of actions, as well as to plan the needed actions and recourses in advance. *Reacting,* re-establishing the operability of the machinery after sudden failures, expected breakdowns of non-critical equipment, or exceptional findings in periodic testing is a critical demand for maintenance. *Monitoring and reflecting* refers to a demand arising from the inherent uncertainties of highly complex systems, and the hence mediated and uncertain nature of the knowledge concerning the object of activity. Reflectivity means critical reviewing of the effectiveness and results of one's actions and of the premises on which the actions are based. Three instrumental demands that facilitate the parallel fulfilment of the critical demands were extracted; *flexibility*, *methodicalness* and *learning*. Working practices necessary for the fulfilment of the OCT are also listed in the Figure.

Main results from the three case studies

The results implied that the three maintenance units emphasised some requirements of the maintenance core task more than others (Figure 107). The emphases also differed between the units. Organisational changes in maintenance activities had also made some features of the task more salient (e.g. emphasis on learning) and put some on the background. A future challenge for all the plants is to organise the maintenance in a way that allows for perceiving and fulfilling all the critical demands of the maintenance task.

Common to all the plants was that at a general level the goals of the maintenance task were perceived to be very clear; maintenance is a prerequisite for the reliable production of electricity. The personnel saw their work as highly important, even though the plants differed significantly on how the personnel saw their work contributing to the overall goals of the organisation. [20] Maintenance work appeared to produce a feeling of meaningfulness when there are technical problems to solve with safety significance and time pressure [19, 20]. This is a paradox in the sense that one of the goals of maintenance is to avoid problems and keep the technology running reliably. If one assumes that the technology in the future can be made more reliable and fewer technical "problems" will occur, then this could be a challenge for the personnel to withhold "meaningfulness" in the sense of the joy felt by fixing problems. Ability to contribute to the perceived core task provides a source of meaningfulness for the employees' work. Maintenance task should thus be focused on maintaining the entire plant, not some individual pump or valve. In other words, we propose that understanding the shared organizational core task should provide meaningfulness in one's work. [20] One possibility for enhancing the meaningfulness of the maintenance work is to try to give the maintenance workers more opportunities to participate in the various modernization projects [2].

The content of the individual jobs is gradually changing in maintenance work. Especially the role of foremen had shifted from participating in the field work to supervision of work from the computer, planning the work and analysing data concerning the machinery (monitoring and reflecting in Figure 107). This change has evoked mixed feelings. Some foremen are afraid of losing the touch to the field work and to their workers and to the machines. However, the enriched job content has been experienced as challenging by some. The current focus on strategic optimization and new information technology can threaten the traditional conception of proficiency (based on handicraft skills and practical experience) among the personnel. The new expectations created by the new technology are not congruent with the old cultural conceptions of a skilled worker. The personnel do not want to see the machinery as merely numbers on a computer screen or data base, but as concrete objects to work and play with [cf. 11, 24].

The results imply that there exists a danger that the routine tasks and the more challenging tasks will be systematically given to different people. Conducting routine tasks is as important for the development of expertise as is conducting the more challenging and unique tasks [16]. Every employee should have the opportunity to do both in an appropriate proportion in order to promote organisational learning and development of competencies. The results gave implications that the competence of the employees was by large taken for granted at the maintenance units. New technology, however, sets new requirements, which means that some of the old habits have to be unlearned. Long tenure as such does not guarantee competence. Long tenure can also lead to routinisation. Experience is then no longer a benefit, but can actually be a source of errors when the work and its outcomes are not actively reflected upon [19]. The aging workforce, the coming generation change, and the new forms of organising work (such as the increased use of subcontractors for special tasks) emphasise the importance of managing these practical challenges.

Conclusions

Maintenance activities have been under various restructuring initiatives. These changes have been experienced as causing stress and uncertainty among the workers. Personnel's experiences are not merely resistance to change; they are also genuine worry about the safety implications of the changes [20]. This may have an effect on the commitment and trust of the personnel toward management. The effect of changes on the reliability of maintenance is not straightforward. Some employees had focused their attention to the social aspects of the work because of unclear organisational goals and structures. The source of the meaningfulness of the work had gradually shifted away from the task itself. Methods for taking the human factors into account in the planning of organisational changes should be further developed.

An important challenge in terms of human reliability in NPPs is the worker's sense of personal responsibility [cf. 3]. In NPPs, the achievement of a sense of personal responsibility is complicated by strict rules, procedures, and a tendency to emphasize shared responsibility and collective action instead of individual initiative [cf. 21, 3, 17]. Results imply that some workers overemphasise the formal structural features of the organization as a source of control and meaningfulness. Responsibility then means that you do what is formally required, not what would be felt personally as a sensible course of action. Effort is thus not directed towards the fulfillment of the organizational core task, but towards the fulfillment of the subtasks and subgoals that the worker is directly accountable for. A challenge for maintenance is to be able to build organisational structures and practices that would facilitate the following conditions: a sense of meaningfulness that is connected to the task itself, a possibility to see the results of one's own work, a realistic sense of control, and a sense of personal responsibility over the plant.

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23. Disseminating tacit knowledge in organisations – preproject (TIMANTTI)

23.1 The role of tacit knowledge and the challenges and methods in sharing it at the Finnish NPPs

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Abstract

The ageing of workforce, the lack of training programs and recruits, and the decline in R&D activities have evoked discussion about the need to preserve nuclear knowledge by transferring it from retiring experts to new recruits. Studies conducted in the nuclear and power industries have found that challenges lie especially in transferring tacit knowledge, which the experts have accumulated through long careers and various experiences in professional settings. This paper examines the role of tacit knowledge and the challenges and methods in sharing it at the Finnish nuclear power plants.

Introduction

Knowledge and the capability of creating and utilizing knowledge have been claimed to become the most important sources of organizations' sustainable competitive advantage these days [1, 2, 3]. Maintaining the expertise and knowledge in organizations is especially topical in the NPP sector as a significant proportion of the human workforce in the nuclear power sector is ageing [4] simultaneously with fewer people studying nuclear science and related fields, with universities giving up their training programs, with R&D activities being cut down and with the amount of new recruits declining [5].

Explicit knowledge and implementation of proper IT systems have been of concern of many organizations recognizing the importance of managing knowledge for safe operation. Data and document management systems have been implemented to capture, store and distribute explicit knowledge. However, recently also awareness of the existence of tacit knowledge in organizations has arisen. This paper explores the role of tacit knowledge and the challenges and methods in sharing it at the Finnish NPPs.

Theoretical background

It has been suggested that knowledge in organizations is built of two types of knowledge: explicit and tacit [6]. Explicit knowledge is defined as the kind of knowledge that can be articulated in a verbal form, be documented and codified, and shared by means of e.g. databases, documents and books. As knowledge management became a topical issue in 1990's, most of the knowledge management efforts focused on managing explicit knowledge, information and data with technology intensive solutions and systems. However, most of the knowledge management projects seemed to fail being not adopted as planned and not ending up with an outcome desired [7]. One explanation for these failures might be the strong emphasis put on only the type of knowledge that was explicit and deliverable via IT systems. To some degree knowledge in organizations has been claimed to be tacit and embedded in persons, involving mental models, crafts, skills, intuitions, hunches, attitudes and feelings which may be very difficult or even impossible to articulate [8]. Tacit knowledge is the practical know-how of individuals, which has been seriously acknowledged as part of knowledge management efforts only recently [9]. Unlike explicit knowledge, tacit knowledge resides in the networks of human beings [10]. Tacit knowledge is not usually acquired in a verbal form and therefore it is at best difficult to share it verbally. It may be shared e.g. through observation, imitation and practice [11], shared experiences [8] and joint activities like being together and spending time together in physical proximity [12].

Whilst the awareness of the role of tacit knowledge in organizations has grown, its diffusion in organizations has proved to be difficult. One strategy has been to explicate [2] the tacit knowledge and share, then, the explicit knowledge. However, the experienced workers usually lack skills, motivation and time to document their expertise [13]. In some companies, technical writers have been recruited to interview the experts and write the documentation [13]. However, problems have still remained as experts have not found time for interviews and sharing the expertise has not succeeded without trust and close relationship between the writer and the expert [13]. It has also been found that essential aspects of tacit knowledge are lost when explicating it [14].

Another possibility would be to share the tacit knowledge in tacit form with all its unconscious and automated elements. From this perspective, e.g. communities of practice [15], mentoring and story-telling [16] have been suggested as proper methods for sharing the complex tacit knowledge. In the NPP context, human resource policies such as internal training through enterprise universities, tutorage of young scientists by seniors, international mobility of workers [4], teamwork, meetings, on-the-job training, site visits, cross-training, shift changes and peer-to-peer communication [17] have been seen as potential solutions for preserving nuclear knowledge.

Enhancing the process of sharing tacit knowledge in tacit form has been suggested to be valuable [18] but challenging. Due to the characteristics of tacit knowledge it cannot be shared through formal procedures but needs motivation and commitment in an environment of trust and care [19]. Due to the special nature of tacit knowledge many challenges in sharing it have been identified: difficulties in perception of one's tacit knowledge; different language between masters and novices; time required for assimilating the tacit knowledge; value not given to aspects of tacit knowledge like intuition; and distance inhibiting the sharing of tacit knowledge as face-to-face interaction is considered critical [9]. Accordingly, tacit knowledge requires the existence of 'strong ties' in an organization in order to be diffused [20], that is, regular contacts between two individuals characterized by friendship and reciprocal favors [21].

Unlike explicit knowledge tacit knowledge cannot be stored in databases but is embedded in people and taken away with them as they leave the company [22]. As a consequence, diffusion of tacit knowledge in organizations becomes especially essential when tackling the issue of employee turnover [22] e.g. due to retirement.

Material and methods of the study

This paper presents results of a qualitative case study in the context of Finnish NPPs in Olkiluoto and Loviisa. Both of the power plants have been established in the late 1970's and their turnover of workers has been low. Currently, a significant proportion of the personnel of NPPs is beginning to retire, which has brought about an urgent need to find out ways to ensure that the loss of know-how due to the retirement would be minimal.

The aim of the study was to describe and understand the role of tacit knowledge at the NPPs as well as the challenges and existing methods in sharing tacit knowledge. The data were gathered by conducting altogether 17 thematic interviews and 4 group discussions. The data were analyzed with a content analysis approach.

Results

Even though some of the interviewed key informants considered the emphasis of the knowledge in the NPP context being on explicit knowledge, also **the role of tacit knowledge** was considered significant. As a consequence of accumulated experiences during the work history, the employees had developed into experts in their own area and this expertise could not be captured in instructions or documents. The experts knew from very little hints the right course of action and the consequences of their actions. It was considered impossible to document all the hints and their possible combinations together with all the potential consequences of each operation, especially as the experts were not even themselves necessarily explicitly aware of their problem-solving.

The role of tacit knowledge was seen especially emphasized in knowledge related to certain issues and processes, e.g. the building process of the power plant (e.g. design specifications and project know-how); the experience of using the NPPs; interpreting and evaluating issues and situations; finding the relevant documentation from the paper files; the know-how embedded in commissioning as well as public approval; domestic and international relations; and fuel acquisition and radioactive waste management. The role of tacit knowledge in the NPP context was considered critical for three reasons: 1) the nuclear technology is remarkably complex; 2) nuclear know-how is only in hands of a few; and 3) the NPP context is safety-critical.

The interviewees identified many **challenges in sharing tacit knowledge** at the NPPs. As the most significant challenge was seen the forthcoming retirement of a large proportion of staff. These employees had tacit knowledge related to e.g. the commissioning and initial operations of the plant, vast experience in operating the plant and effective domestic and international relations. At both the NPPs the responsibilities had been divided between the employees and it was common that knowledge of a certain issue was in hands of only one or at most a few persons. This had not earlier caused problems due to the low employee turnover but now the challenge was to transfer this know-how to the new employees. Accordingly, the interviewees wished that the existing practice of specializing in certain tasks and responsibilities would not be maintained and that the new employees would develop a more multifaceted base of expertise.

Apprenticeship was seen as a way to share the know-how but it was considered taking rather a long time, for example two years of working together. This raised a question of finding resources to recruit the apprentices and ensuring the motivation of the apprentice to follow the master for few years. The newcomers were given status and responsibilities by transferring the experts to other devisions, but the connection between the ‘master’ and the ‘apprentice’ was similarly reduced. Also, there were differences in the ability of the experts to guide the newcomers and share their expertise. The know-how of the experts was self-evident for themselves and at least some experts had trouble in taking a stand of a novice.

Some of the know-how of the experts was considered such that it could have been documented. However, at least some documents would be long and complex and it was suspected that no one would absorb this kind of knowledge by reading instructions. Also, not all of the tacit knowledge of the experts was considered worth transferring: especially some practices were found more effective among the new generation.

At the other NPP in Finland, a challenge related to the tacit knowledge was also the building of a new nuclear plant and documenting the knowledge related to the process in order for it to be as much in explicit form as possible and for the transfer of the remaining tacit knowledge to be effectively controlled and planned. Furthermore, challenges were also found in creating new training material and developing more multifaceted training, which would lead also to effective transfer of tacit knowledge.

Overall, six existing **methods for sharing tacit knowledge** were utilized in the organizations. Three of the knowledge sharing methods were connected to sharing tacit knowledge in a tacit form. *Mentoring* had been used to systematically socialize a new worker to the organization. New recruits had been assigned a mentor, with whom he could discuss his work and the problems encountered in work situations. The mentoring relationship lasted one year with the mentor and the new worker meeting on a monthly basis. Mentoring had been in some cases effective but in others less effective, depending on the quality of the mentoring relationship.

Also *apprenticeship* was utilized in sharing the expertise of employees about to retire. New employees had been recruited and given responsibility whilst the aged experts were still available to give guidance. At both the plants the early recruitment of new workers was emphasized as the knowledge to be shared was complex and sharing it took time, especially in maintenance, in which it was in some cases essential that the expert and new worker were able to work together during at least one or two revisions. Apprenticeship was used as a 6-month training period during the education of new operators. The expert and the novice worked together in a shift and the new worker was able to pose questions in problematic situations. In Olkiluoto, the new worker performed work tasks independently and, when needed, asked the expert for advice and guidance. In more extensive and significant tasks, both the expert and the novice participated. At both the NPPs, individual differences had been noticed in the motivation and ability of the experts to guide the new workers and to share their knowledge and responsibilities. There was a need to systematize the apprenticeship as even though master-apprentice pairs had been arranged it was not known what actually happens during the apprenticeship period and how tacit knowledge is shared.

Tacit knowledge was also shared through *occupational instruction*. In Loviisa, the new worker received occupational instruction during his first year of employment. The new recruit was assigned to perform various, pre-defined work tasks with experienced workers first only observing and gradually receiving more responsibility in performing the task. All the experienced workers acted as "trainers" for the new workers. Although very useful, occupational instruction had not always produced desired results. The main challenge was enhancing multi-professionalism instead of specialization: even though the new workers learned all the required competencies, some experienced workers wanted to specialize in some specific tasks leaving the new recruits performing only some limited tasks. This had in some cases led to experienced workers being overly stressed by their work and new workers being unmotivated by their simple tasks.

In addition to methods by which tacit knowledge was shared in tacit form, there were also methods for sharing tacit knowledge by explicating it. *The writing of memos* meant that employees about to retire wrote manuals and guidebooks, in which they gathered the most relevant and critical knowledge in terms of their work. These documents had been used by some co-workers, who found that some memos were informative and useful but it was difficult to understand some of them. The interviewees saw that for some experienced workers, it was difficult to assume the role of the novice and to know what kind of knowledge should be shared and how. Tacit knowledge was also self-evident for the experts. The interviewees saw that all knowledge cannot be explicated and, thus, tacit knowledge plays always a role in work situations, to some extent at least. Also, *situation reports* were made in deviant work situations. In the reports, an employee described and explained the causes for and the consequences of the situation. In this process, also tacit knowledge embodied in the situation was partly explicated and brought to a verbal form. Finally, *training materials* had been produced in co-operation between an expert and a novice, with the expert knowing the content and the novice posing questions and ideas. In compiling the material, tacit knowledge of the expert had been shared with the novice and partly explicated in the training material. Producing training materials together had been motivating for both parties as they both trained their colleagues or other professionals in the organization with the produced material.

Conclusions

Even though the requirements for documentation at NPPs are high, the role of tacit knowledge seems to be essential for effective operation at NPPs. Tacit knowledge was seen to result from gaining experience of working at the NPPs and was considered especially significant due to the safety-critical nature of NPP operations. The greatest challenge related to the transfer of tacit knowledge was seen the forthcoming extensive retirements. Documenting the tacit knowledge was considered worthwhile in certain tasks. However, tacit knowledge was seen as complex and intuitive, and even though it could in some cases be documented, it was assumed that it would be impossible to learn the knowledge from the complex documents. There were also several obstacles for externalizing tacit knowledge: the experienced workers viewed their expertise as self-evident and were unable to reflect their tacit knowledge without someone posing questions; it was difficult for the experts to assume the position of a novice worker; novice workers were not able to understand the documents made as the experts and novices did not share the same backgrounds; the novice workers could not pose questions if knowledge was shared in an explicit form; and it was difficult for the novices to practically apply knowledge obtained from documents. The interviewees emphasized that tacit knowledge cannot be learned from lectures or documents but builds up in face-to-face interaction in everyday work situations.

In social interaction, psychological and social issues come into play. Motivation and ability to share tacit knowledge were especially highlighted when tacit knowledge was shared in tacit form. The eagerness of the new workers to learn and the enthusiasm and willingness of the experts to share their tacit knowledge and responsibilities seemed to be determinants of the successfulness of sharing tacit knowledge. Also the ability of the experts to reflect on and communicate their know-how, and the ability of the learners to pose questions seemed to affect effectiveness of tacit knowledge sharing.

Even though several methods for sharing tacit knowledge were used in the organizations, tacit knowledge sharing occurred fairly unsystematically. The representatives of the organizations knew that tacit knowledge was indeed shared, however, it was not known *what actually happened* when an expert and a new worker interact and *how tacit knowledge was shared* in this relationship. The NPPs and their subcontractors with ageing human workforce need further research on the complex mechanisms of sharing tacit knowledge to be able to preserve and diffuse the expertise. Specialization was also seen as a problem as the experts were retiring and it was wished that the new workers would build wider and more multifaceted know-how.

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24. Potential of fire spread (POTFIS)

24.1 POTFIS summary report

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Abstract

The strategic goal of POTFIS project and its background philosophy is outlined, and two new examples as parts of this development are described: CFD simulation of fires in turbine halls, and Monte Carlo calculation of large and complicated fire scenarios. For the former guidance is given how to avoid flow oscillations, which although physically based phenomena, are numerical artefacts caused by simplified geometry. For the latter an economic Two Model Monte Carlo method is developed allowing use of CFD at a needed accuracy but fraction of cost of full one model Monte Carlo.

Introduction

Internationally the fire safety part of PSA in NPPs is not at the same scientific and technical level than most of the other parts of PSA. First US Nuclear Regulatory Comission's (NRC) paper [1] coined to three options the improving discussions during 1990's to 'pursue to develop a rulemaking for transitioning to a more risk-informed, performance-based structure for fire protection regulation of nuclear power plants': (1) develop a performance-based, risk-informed fire protection regulation, (2) develop a performance-based, risk-informed consensus standard, or (3) maintain the existing fire protection regulations and guidance. The NRC policy chosen was option 2 as a result of development of a standard [2]. An implementation proposal as a new consensus guideline was released recently for wider comments [3].

In Finland STUK asked for the goal '… to transit to more risk informed …', but left the alternatives to scientists. Finnish research program FINNUS (1999–2002) set 1998 fire research goal: '…fast risk analysis methods … will be developed.' [4, p. 13]. Thus already 1999 the first prototype version of our Monte Carlo calculation platform PFS was presented [4, Appendix 1, p. 23], which was later expanded to a general calculation platform [5]. Encouraged by success in creating stochastic framework for fire-PSA, Keski-Rahkonen wrote 2002 in plans of the next program SAFIR (2003–06): 'The major strategic problem during SAFIR is the ability to predict potential of fire spread in given scenarios', [6, p. 55]. From discussions with authorities and utilities a conclusion was made, the most demanding scenarios influencing CFD in our existing plants are cable fires in places containing two or more redundancies without partitions. Although the consensus guide [3], not available during the planning of SAFIR, contains new information on some of the proposed items, the major goalsetting holds, and is closer to option 1 mentioned above and thus mainly somewhat ahead of those in [3].

Main objectives

The strategic goal of POTFIS project is quantification of fire PSA tools. Deterministic calculation of fire consequences using appropriate CFD tools is already satisfactory, if fire size can be given as input. However, lack of tools to predict fire size on extended solids, fire spread velocity especially on cables, is the biggest strategic missing link of the models. 'The tactical objective of this project is to carry out interactively modelling, numerical simulation and experimental work on the relevant, most promising simple solid fuel scenarios for qualification and validation of specific models. For economizing the effort parametric studies will be carried on smallest possible sample sizes. Scaling up to real size will be made subsequently to demonstrate the validity of proposed models', [6, p. 55].

Three subprojects were started: (1) flame spread experiments on cables, (2) fire spread modelling, and (3) reliability of active and operative fire protection. Item 1 consists of screening experiments at various scales on vertical flame spread on and autoignition of thin solids. Item 2 consists of literature review and theoretical modelling of vertical flame spread, development of Monte Carlo techniques to allow use of deterministic CFD models like NIST's LES model FDS for complicated scenarios, and participation in International Collaborative Project to Evaluate Fire Models for Nuclear Power Plant Applications (ICP). Item 3 included completing the modelling effort of active fire protection devices started during FINNUS [4].

In this interim report early results are presented here, but details on fire safety on cables, and vertical flame spread are found in the accompanied special report.

Simulation of fires in turbine halls: horizontal vents

As part of comparison of fire simulation programs a subtask of ICP was a benchmark calculation of turbine hall in a NPP. The typical scenario is given in Figure 108d: large open halls on top each other connected with small horizontal openings. This is a challenging problem, because flow through horizontal vents might be unstable. Although in Finnish applications CFD models are standard tools in such applications, this is not the case in most of countries. Therefore, also various zone models were included. CFAST code, the most common of them, use Cooper correlation [7] for flows through horizontal vents.

During the first round two problems aroused: (1) many other zone models did not have tools for horizontal vents, and (2) also CFAST results seemed to be in error as compared with more reliable CFD simulations. For problem (1) it was reminded that for modelling a horizontal vent, a simple approximate way, Rockett proposal, can be applied as explained in Figures 108a and 108b, and described in [8]. To compare performance of Rockett proposal with Cooper correlation, and well as practical utility of the models by novel users, five students were given tasks to simulate blindly by CFAST horizontal flows in configuration of Figure 108d. After a coding error detected during the first round was corrected in CFAST, both methods gave equivalent results, which were also comparable with those obtained from CFD calculations as shown in Figure 108c, [9].

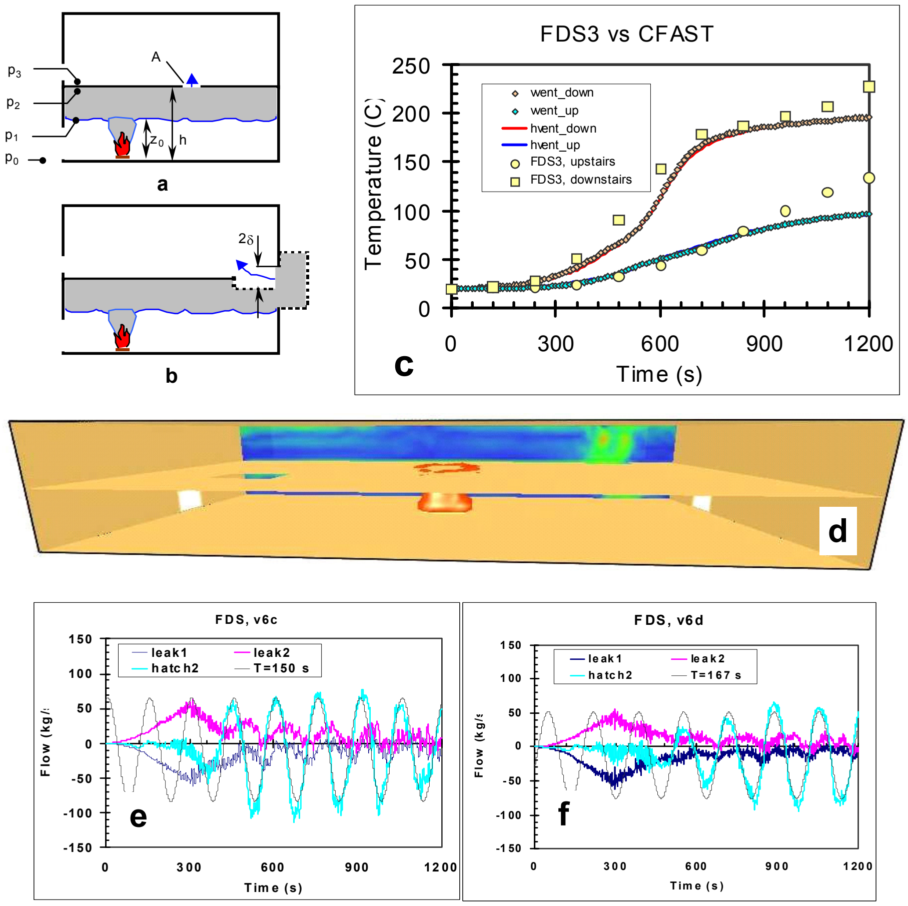

Figure 108. (a) Unstable flow through horizontal vent replaced by (b) flow through a virtual vertical vent, [8]. (c) Both zone model methods and CFD yield approximately the same results (d) in a turbine hall configuration, but, (e) and (f), CFD calculation could also show 'spurious' oscillatory behaviour of flows, [9].

Using CFD simulations internal oscillations of air flow through the vents were observed in some cases as shown in Figures 108e and 108f. The source of those oscillations was not clear, but when convergence and cell size effects were screened to be unlikely, analogous physical processes were also inspected, [10]. Possible phenomena were (a) cellular standing waves, (b) internal seiches in lakes (or the ocean) with a well-mixed warm surface layer separated from a colder homogenous region by a shallow thermocline, or (c) layer of high gradient stratified fluid, Brunt-Väisälä phenomenon. All these cause oscillations at discrete frequencies, but quantitative comparison is still underway [9].

Monte Carlo platform extended to use also CFD models

Our earlier Probabilistic Fire Simulator (PFS) [11] is extended to perform a Monte Carlo simulation using different fire models, including CFAST two-zone model and FDS fluid dynamics model. In this work, a new technique is developed for the use of two different fire models in the same Monte Carlo [12, 13]. The Two-Model Monte Carlo technique provides a computationally effective means to improve the accuracy of the fast but inaccurate models, using the results of the accurate but computationally more demanding models. The technique is tested in three scenarios: approximation of an analytical function, calculation of a ceiling jet temperature and a simulation of a simple room fire (Figure 109).

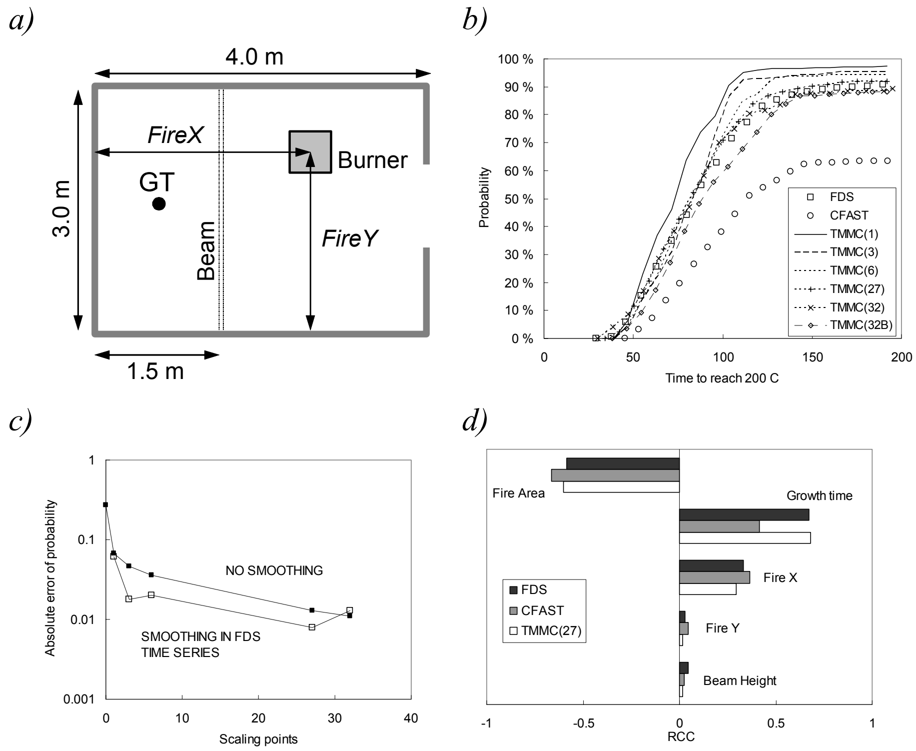

Figure 109. (a) The geometry of the room fire scenario: GT shows the location of the heat detector and the gas temperature measurement point 5 cm under the ceiling. (b) A comparison of predicted probability distributions of time to reach 200 °C. (c) The accuracy of predicted final probability as a function of number of scaling points. (d) Rank order correlation coefficients for the gas temperature reaching time.

Applications

CFD calculations of turbine hall fires are needed right now for Olkiluoto 3. Since oscillations in real life are unlikely, and are here products of extreme simplifications of the geometry for simulations, practical application to avoid these 'disturbances' is to divide large compartments to smaller subcompartments to generate attenuation.

Monte Carlo calculation of fires in switchgear rooms and cable tunnels is going on for spaces taken from our NPPs.

Conclusions

As a part of wider strategic research to carry out fire-PSA quantitatively some parts of it have been finished. A new phenomenon, large scale collective flow oscillation was observed, and guidance is given to avoid that numerical artefact in real simulations. Two model Monte Carlo was demonstrated, which is able to use CFD-calculation in real fire scenarios at a fraction of the cost of full one model calculations. As a result routine engineering applications of Monte Carlo even in most complicated fire scenarios becomes possible.

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24.2 Experiments and modelling on vertical flame spread

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Abstract

The principle and some preliminary results are shown of a new vertical flame spread modelling effort. Quick experimental screenings on relevant phenomena are made, some models are evaluated, and a new set of needed measuring instruments is proposed. Finally a single example of FRNC cable is shown as application of the methods.

Introduction

The strategic goal of POTFIS project is quantification of fire PSA tools. Deterministic calculation of fire consequences using appropriate CFD tools is already satisfactory, if fire size can be given as input. Lack of tools to predict fire size on extended solids, fire spread velocity especially on cables, is the biggest strategic missing link of the models. Literature and studies indicated significant international progress during recent years in understanding flame spread [1]. Small scale experimental efforts were concentrated to continue to test asymptotic flame spread models of single vertical objects, which is geometrically a simple configuration but important to fire spread. The tactical objective of this project is to carry out interactively modelling, numerical simulation and experimental work on the simple solid fuel scenarios for qualification and validation of specific models. For economizing the effort parametric studies and screening experiments will be carried on smallest possible sample sizes. Scaling up to real size will be made later during the program to demonstrate the validity of proposed models.

For assessing the probability of the risk of cable fires in a given space in a NPP the whole spectrum of the ignition sources must be covered instead of a single or a few ‘design fires’ as is done traditionally. Therefore, also the reaction of the target has to be mapped from the whole spectrum of exposures possible. For fire size on solid fuels two factors count: (i) rate of heat release (RHR) per fuel surface, and (ii) the surface area of the burning solid. Item (i) is well known and depends mostly on the material. Cone calorimeter is a viable tool to measure the needed parameters [2]. Item (ii) is a dynamic variable, and only partially known. It is determined by flame spread on surface, which includes a number of different modes. For fastest of them, upward spread in concurrent turbulent flow, no general models are available. For estimating fire consequences in a NPP, calculating the area of the fire is the key task. Here we try an engineering approach on the problem [1].

Simple isolated, and partial results are shown, which at this moment are developed and analysed, but they are all parts of the same larger scheme not yeast ready for presentation. As major sample material various pieces of wood were used although our target is cables. The selection is motivated because various wood materials are quickly available and it can be worked easily into needed forms. Wood as a charring material is a good model for cable insulation material. Finally experiments on flame spread and ignition of alumina trihydrite (ATH) flame protected cables were measured and are shown as a timely application of the method.

Experiments on vertical flame spread and ignition

Bench-scale experiments on wood sticks and cables have been carried out to test measuring techniques as well as to screen physical phenomena and roughly select models. The experiments are grouped as follows, with the main focus shortly mentioned

1. flame spread experiments on 2 m long wood samples W1–W17
   samples differing in diameter (4...21 mm), check of measuring techniques, verifying constant asymptotic flame velocity, screening thickness effects
2. flame spread experiments on 0.3 m long wood samples S1–S22
   samples differing in density (220...1020 kg/m^3), screening effect of density on various burning processes as well as influence of possible differences of wood from different species
3. flame spread experiments on 0.3 m long pre-heated cable and wood samples F1–F31
   samples differing in pre-heating (20...340 ^oC) and density (400...780 kg/m^3 for wood samples) exploring effect of energy balance on flame spread velocity
4. flame spread experiments with a modified cone calorimeter on wood samples C1–C14
   samples differing in density (120...640 kg/m^3) and irradiation conditions (32...72 kW/m^2), suitability of modified system to measure parameters of flame spread using exact adiabatic boundary condition of sample backing
5. autoignition experiments on cable and wood samples A1–A15
   wood samples differing in density (420...800 kg/m^3) exploring temperature limits and time scales for non-piloted ignition
6. heating experiments on wood samples H1–H4
   effect of moisture, determination of time scales for rate of temperature rise as well as validating heat transfer models for wood samples

Experiments in groups 1–3 were made with samples supported from their upper end in a test rig, and ignited from below with a small propane gas burner as presented in Figure 110a for experiments in groups 2 and 3. The longer samples in group 1 were attached similarly in a higher test rig with somewhat differing thermocouple instrumentation.

Flame spread was monitored with thermocouples in the vicinity of the sample surface, by visual observations and taking digital photographs.

The experiments in group 4 were made using a standard cone calorimeter [2] modified by turning the radiation cone upside down and placing a vertical sample supported from its upper end in the centre of the cone as shown in Figure 110b. The sample ignited spontaneously at higher irradiation levels, and was ignited with a match at lower irradiation levels.

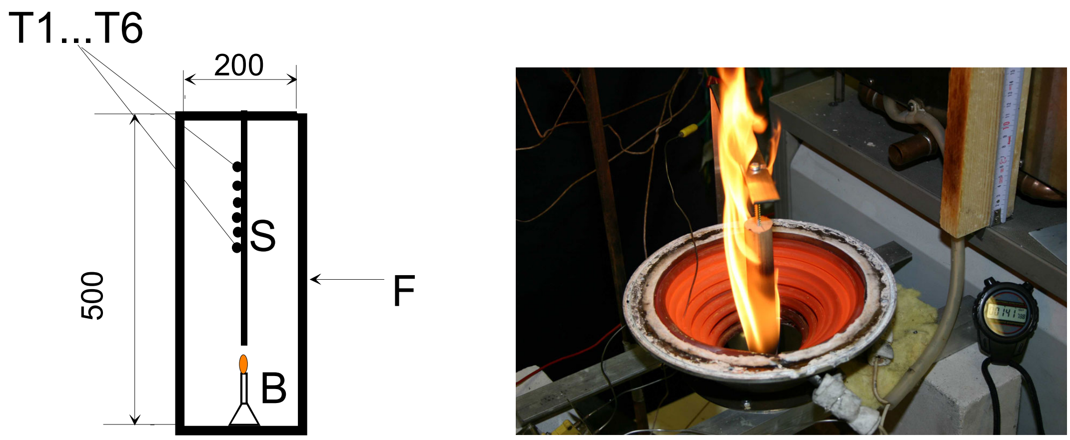

Figure 110. (a) Experimental set-up for vertical flame spread experiments of 300 mm cables and wood sticks differing in density and pre-heating temperature: S wood or cable specimen, F supporting frame, B Bunsen burner, T1–T6 thermocouples at 25 mm vertical intervals; dimensions in mm. (b) Modified cone calorimeter [2] experiment, pine sample, 3 seconds after spontaneous ignition.

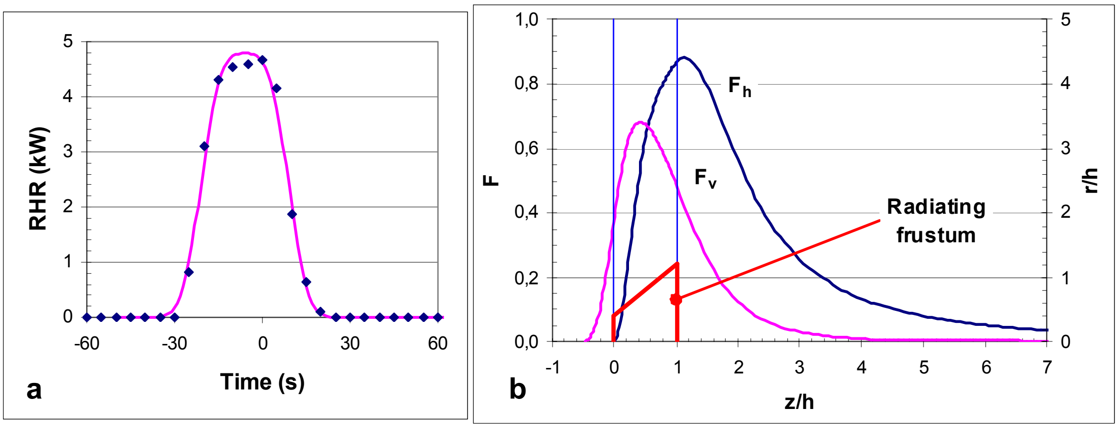

Figure 111. (a) Response of cone calorimeter to a 30 s long RHR pulse (dots) and a theoretical fit (line) using 5 s standard deviation Gaussian window. (b) View factors F from a frustum on the horizontal (F_h) and vertical (F_v) surface element on the vertical axis of a cone as a function of nondimensional distance z/h. The radiating inner surface of the frustum is depicted as a heavy line in the interval 0 < z/h < 1.

Thermal radiation field close to the frustum of the cone calorimeter was modelled by deriving analytical calculation forms as a function of dimensions. In Figure 111b the view factor of the frustum is plotted as a function of distance for a surface perpendicular to (Fh) or parallel with (Fv) the axis of the cone. The model showed the radiation field is so uneven along the axis, that experiments using vertical samples as shown in Figure 110b are not feasible. Still experiments C1–C14 showed as indicated in Figure 112, that the problem caused by ill-posed background heat transfer, was removed using vertical samples. In the future, a revised calorimeter heater is needed for determination of energy balance parameters: either a longer cylinder around a vertical stick or two parallel plates on either sides of a planar sample are needed.

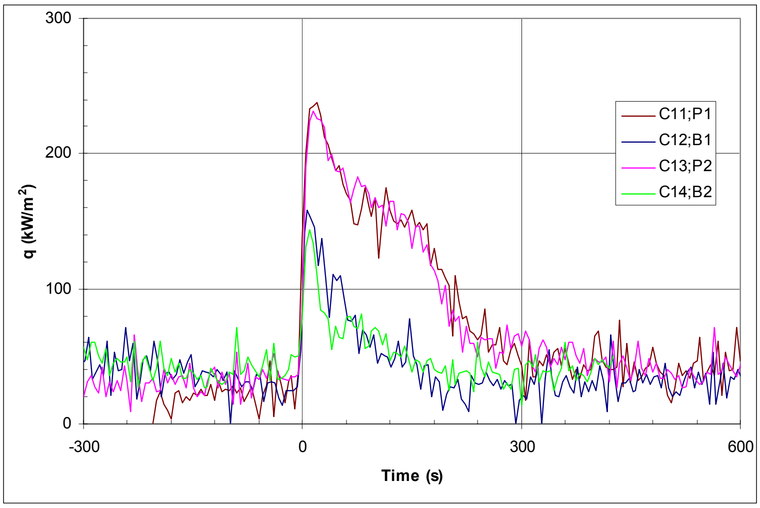

Figure 112. RHR from cone calorimeter experiments C11–C14 using vertical cylindrical samples (ϕ 20x140 mm) on the axis: P1 & P2 pine, B1 & B2 balsa at two irradiation levels. Time scales were adjusted by placing the origin at the inflexion point of ignition.

The time resolution of the RHR measurement by cone calorimeter was determined by placing a gas flame of constant size under the cone for short times (1–60 s) and then removing it quickly. The dots on Figure 111a indicate obtained response for 30 s pulse. The full line in Figure 111a is a convolution of the rectangular pulse with a Gaussian window of σ = 5 s, which gave a reasonable fit for all pulses measured. This allows estimating the distortion caused by the instrument in plots like shown in Figure 112.

For modelling flame spread, a simplified engineering approach is used, where ***on the average and in a coordinate system moving with the flame front*** (Figure 113a) a stationary flame is observed. A very simple heat transfer model was written presuming temperature on sample surface is proportional to the view factor the flame shows from the observation point. In the moving reference system locked to pyrolyzing front the flame ABC is a planar thin sheet of infinite width but finite length at some standoff distance from the surface of the sample. It is viewed from a position of a thermocouple at point P in Figure 113a. Assuming the flame sheet to shrink into a surface and extending to infinity in a plane perpendicular to paper, calculating the view factor becomes a problem of plane geometry.

The intermittent part of the turbulent flame sheet is modelled by the section BC which is on the average at an angle φ relative to the surface of the sample. The length of its projection on the surface is s. For normalization of the length scales the pyrolyzation length was used. Photographs of flames on pine sticks (W15) in Figure 113b demonstrate this thinking is viable.

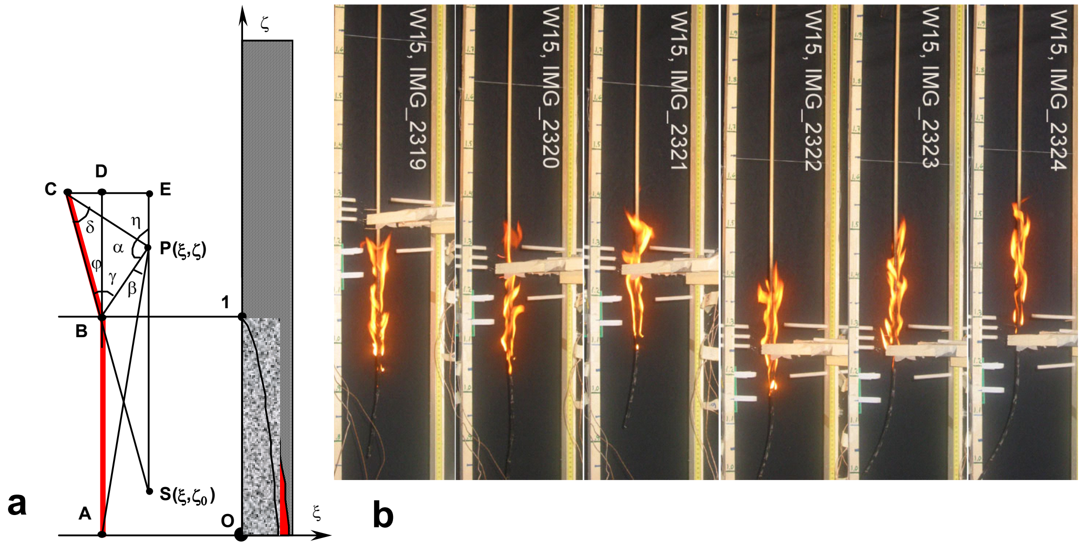

Figure 113. (a) Stationary flame sheet on moving co-ordinate system locked to burnthrough edge O of the sample. (b) Photographs on flame propagation on a vertical pine stick of experiment W15.

The agreement between this simple model and observations from literature [3] is remarkable (Figure 114a). This gave the motivation to pursue describing the flame as a flame sheet of constant shape and position relative to the surface but moving with constant speed up the sample. For cylindrical samples plane geometry is replaced with cylindrical geometry. Mathematics is somewhat more complicated, but can be carried out analytically in closed form as shown for modelling cone calorimeters curves of Figure 111b. The main conclusions are still the same as for the planar model of Figure 113a. This was demonstrated by measurements shown in Figure 114b in experiment W10 using a pine stick of 7.1 mm diameter. Temperature curves look qualitatively very similar; quantitative evaluation will follow.

In series W1–W17 velocities of flame spread were measured on vertical 2 m long rods of differed diameters. In Figure 115 results of experiment W15 are shown.

Burnthrough (BT), pyrolysis front (PF) and flame tip (FT) are recorded visually (open symbols) and photographically (filled symbols). Linear lines (solid lines) are fitted at longer times to photographic observations. There might be initially some transients which smooth out inside 0.5 m from ignition. Figure 115 yields both constant asymptotic flame front velocity, and pyrolyzing length (AB in Figure 113a). There are some quantitative differences between photographic and visual observations, the meaning of which will be clarified later.

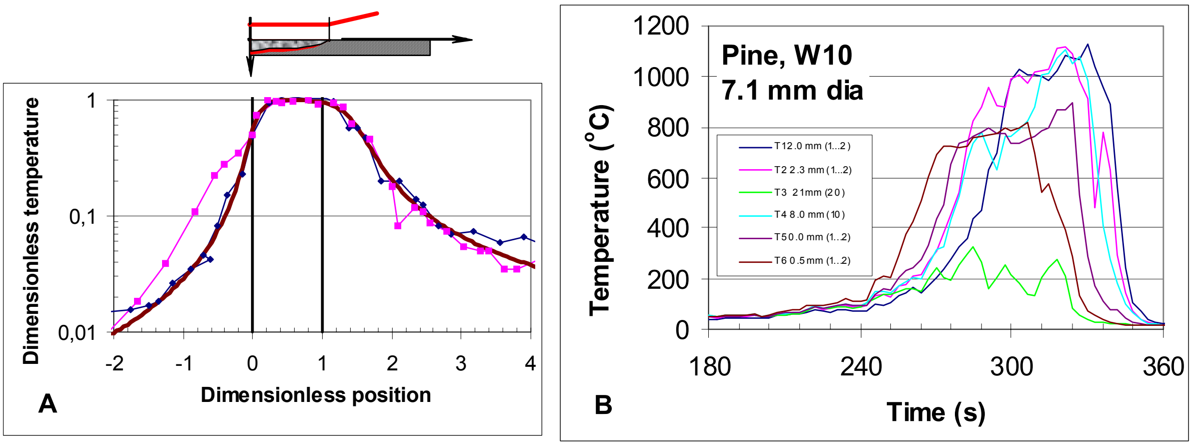

Figure 114. (A) Dimensionless temperature as a function of dimensionless position (heavy line) as fitted to data (squares and diamonds) by [3]. (B) Temperatures T1–T6 in the gas phase as a function of time in Experiment W10. Notice that equivalent metrics, time scale and dimensionless position run in opposite directions. Therefore figures A and B are mirror images of each other.

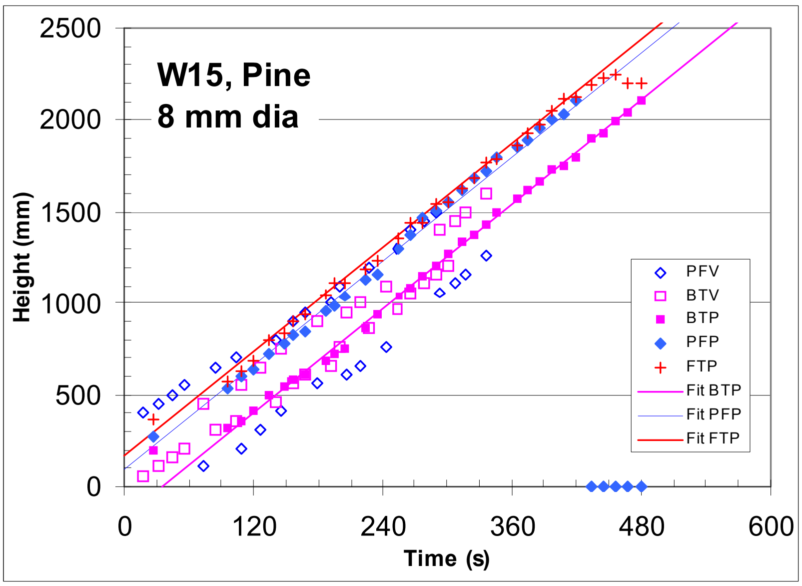

Figure 115. Flame spread as a function of time in Experiment W15.

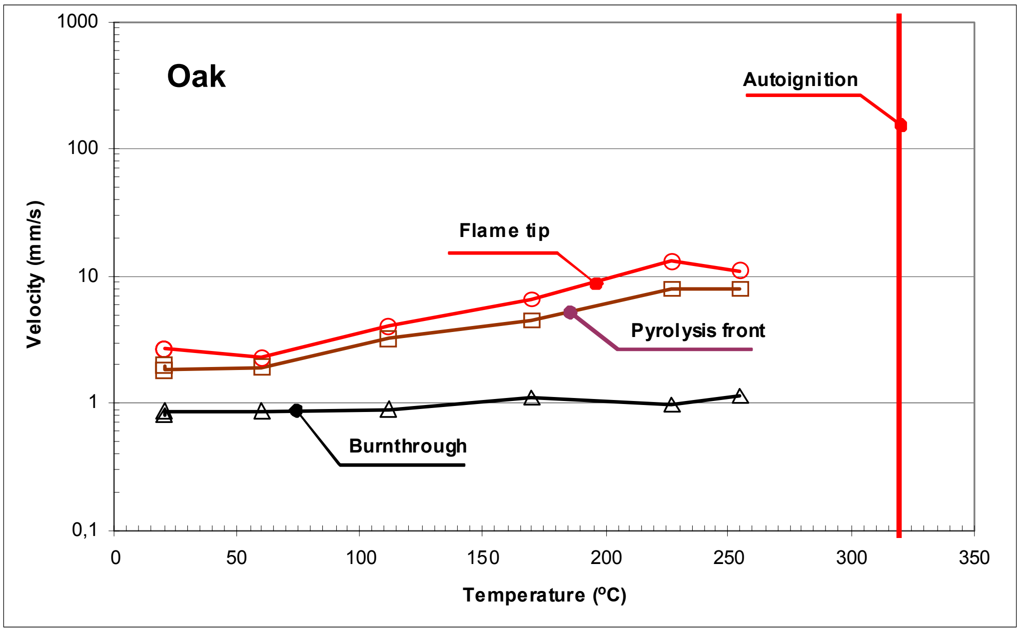

Figure 116. Velocity of the burnthrough, pyrolysis front, and flame tip on vertical oak sticks as a function of sample temperature.

Velocities of burnthrough, pyrolysis front, and flame tip were measured quickly in series C1–C14 as a function of the initial temperature of the sample as shown in Figure 116 for oak. It is shown, that velocities increase roughly exponentially with temperature until close to autoignition temperature.

The cable or wood sample was heated in a laboratory furnace with slowly rising temperature until the sample ignited. Furnace and sample temperatures were monitored with thermocouples. The time-temperature curves for experiments in series A1–A15 on oak (A11) and a fire retardant non-corrosive cable (FRNC) sample (A4) are shown in Figure 117 for example. In addition of some cable samples autoignition temperatures, and time scales were extracted systematically for obeche, pine and oak representing low, average and large densities of available woods.

FRNC sample ignited at 394°C (Figure 117a), and burned totally after that. The results show that the flame retardant properties diminish with increasing temperature until finally ignition occurs without external pilot. In FRNC cables flame retardancy is based on endotermic dehydration in alumina trihydrate (ATH), [4]. When temperature rises, various processes lead to decrease of net water as shown by differential thermal analysis (DTA) and thermogravimetry (TGA) in Figure 118c. Even in room environment wood contains some 10% moisture (Figure 118a), which has a clear effect in heating as compared to oven dry wood (Figure 118b). Heating experiments on wood samples H1...H4 were carried out where the sample, originally in ambient temperature, was rapidly inserted in a hot furnace. Heating of the wood sample and effect of moisture was monitored through temperature measurements inside the sample and in the furnace.

Delay in temperature rise seen around 500 s in Figure 118a is a similar phenomenon at lower level than flame retardancy in FRNC cables using ATH. A new test rig about 2 m long for determining the temperature limits where the flame retardant properties have decreased to a critical level is proposed and will be built.

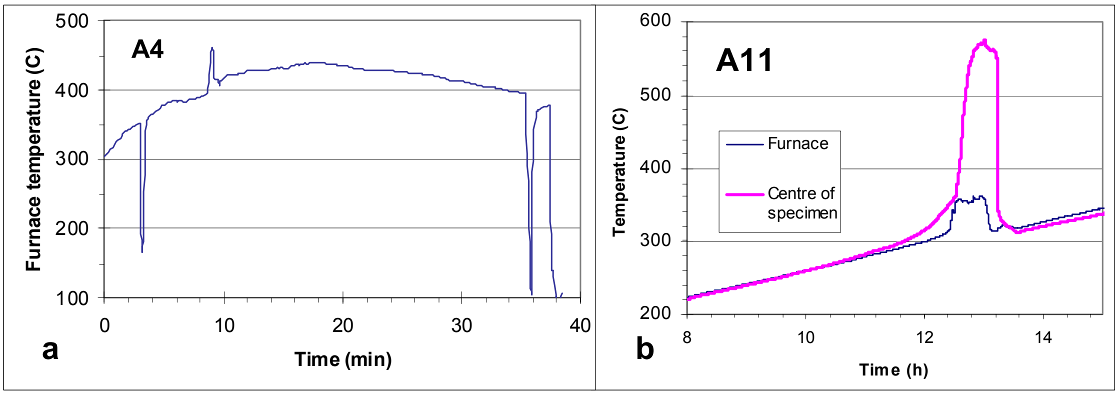

Figure 117. (a) Autoignition experiment A4 of flame retardant non-corrosive cable Afumex 500 NHXMH-J sample at 394°C around 10 min. The temperature drop at 3 min. comes from opening the furnace door for a fast visual check of the system. (b) Autoignition experiment A11 of oak sample at 313°C around 13 h. Temperatures in the furnace and inside the sample were measured separately.

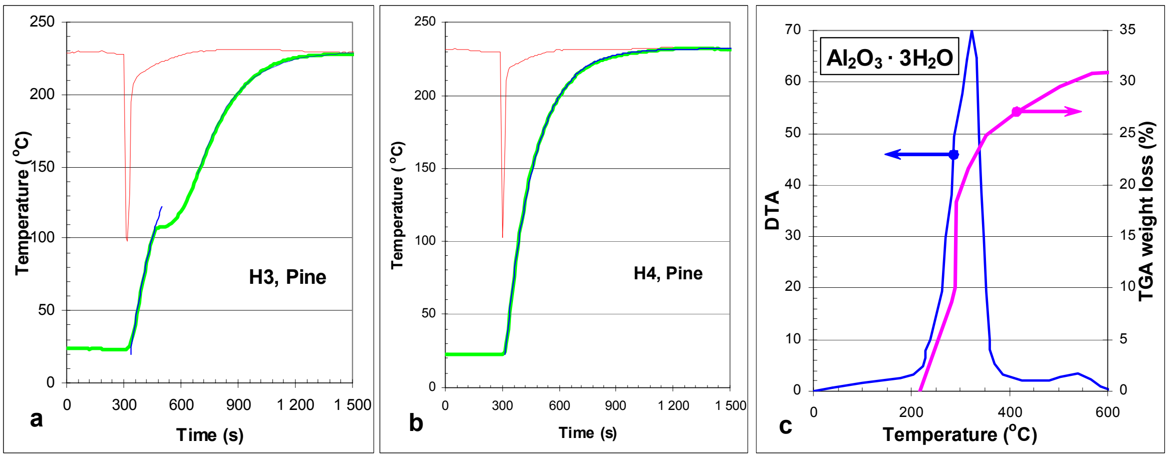

Figure 118. (a) Effect of 9.5% moisture on the heating of pine as compared with (b) oven dry pine. (c) Temperature dependence of removal of water from ATH as measured by DTA and TGA, [4].

Applications

Since FRNC cables are very relevant for the fifth NPP in Finland, samples of respective cables were measured using all the methods described here. Limits of ignition, and potential flame spread could be assessed quantitatively, and were already used preliminarily in real assessments.

Conclusions

A new engineering method to model vertical flame spread on solid surfaces is proposed, and also checked preliminarily using quick small scale screening experiments. A new modified calorimeter test, as well as a new flame spread test rig is proposed and will be build soon in the laboratory. Pilot experiments on a real problem of FRNC cables was carried out since it was already needed for our new NPP evaluations. Final verifications of the proposed method and test equipments remains to be carried out at larger scale once new devices and more elaborate models are available. The new model will be implemented during the later part of SAFIR program into LES fire simulation model FDS [5] allowing calculations, demonstrations and comparisons at larger scale.

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25. Principles and practices of risk-informed safety management (PPRISMA)

25.1 PPRISMA summary report

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Abstract

The PPRISMA project deals with both risk-informed decision making and methods for risk assessment. Approaches have been developed to support maintenance and operability planning, management of fire situations and safety classification. A method for the reliability assessment of computer-based systems has been developed. In addition, the research activities are carried through international co-operation.

Introduction

Risk-informed safety management means use of information from probabilistic safety assessment (PSA) to support decision making in various contexts. In nuclear safety regulation, risk-informed applications are an option to present licensing practices. The Finnish regulatory PSA guide calls for risk-informed assessment of safety classification, in-service-inspection programmes, in-service-test intervals and allowed outage times of equipment. Similarly, maintenance and surveillance programs, training of personnel, working out of procedures and ways of acting can be assessed. PPRISMA project deals with both developments of risk-informed methods and specific PSA modelling issues related to uncertainties that can hinder implementation of risk-informed applications.

Main objectives

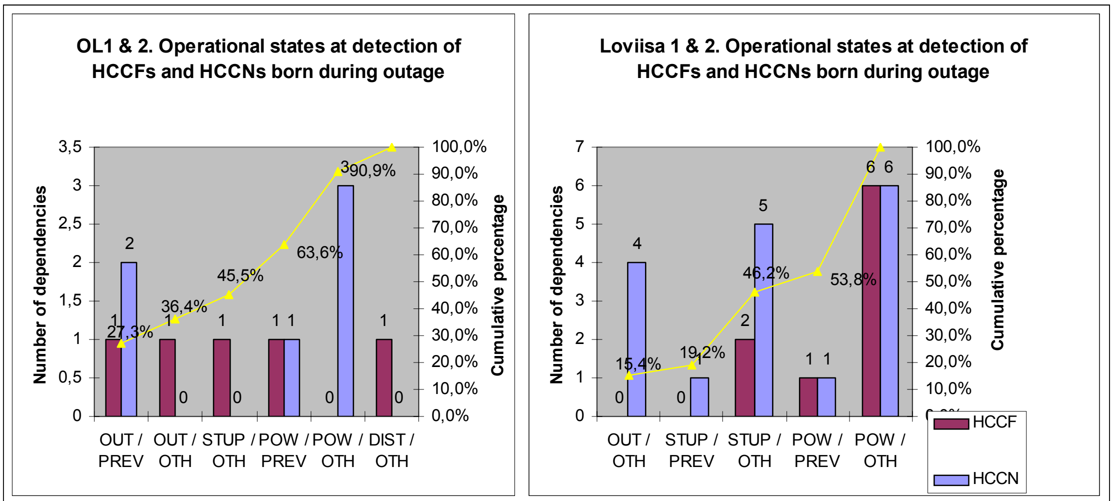
The main objectives are to develop risk-informed decision making methods integrating results from risk and reliability analyses with other expertise in the problem domain, to develop assessment methods for plants' operation and maintenance to enhance planning of activities and acting in situations, to develop methodologies in the problem areas of PSA, to advance skills in risk analysis, to assure the competence transfer to the new generation and to participate in international co-operation.

Risk-informed decision making

A study was made of status of risk-informed decision making at Finnish and Swedish nuclear power plants and safety authorities [1]. In Finland, the recently issued regulatory PSA guide requires the licensee to use the PSA in support of licensing of new NPPs and of resolving safety issues at operating NPPs. The regulatory body STUK has performed several pilot studies to facilitate risk-informed applications. In Sweden, the use of PSA is less mandatory than in Finland. In both countries, the nuclear power plants have applied PSA in many areas. The main problems related to the applications are: quality of PSA, communication between parties involved, and acceptance of risk-informed decision making. The study presents a decision theoretic framework to verify that most important principles have been followed to a reasonable extent.

Maintenance and operability strategies

The focus in human reliability analysis has traditionally been on human performance in disturbance conditions. On the other hand, human maintenance and planning failures and design deficiencies, remained latent in the system, have an impact on the severity of a disturbance, e.g. by disabling safety-related equipment. Especially common cause failures (CCFs) can affect the core damage risk to a significant extent. The topic has been addressed in Finnish studies, where experiences of latent human errors have been searched and analysed in detail from the maintenance history in the Loviisa and Olkiluoto NPPs. The most errors related to maintenance and modifications stem from the refuelling and maintenance outage periods [2]. Figure 119 shows the distribution of the fault detection states of human common cause failures originated from outages. The studies suggest improvements in the planning of installation inspections and testing as well as in the operability verification after work actions.



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

Figure 119. Distribution of fault detection states of human common cause failures originated from maintenance outages. (HCCF = critical human common cause failure, HCCN = non-critical human common cause failure).

Construction of an analysis process model for risk informed maintenance and operability planning is under way [3, 4]. The model covers better such safety and reliability management and economic loss prevention criteria as risk importance, CCF avoidance and operability verification than the present streamlined RCM (reliability centered maintenance) based maintenance analysis and planning methods. This model development was also supported by the first trial study on application of the Burden to Importance Ratio (BIR) for resource allocation of condition based maintenance of a safety related system at the Loviisa plant [5].

Risk-informed ways of management of fire situations

VTT has developed a method for supporting the management of NPP fire situations from the main control room point of view [6–8]. The method has been created by utilising interdisciplinary expert group work, carried out with Loviisa plant. It provides a plant-specific model which is a generic description of the shift supervisor's most important cognitive task demands during fire situations and the recognised prerequisites for fulfilling these demands. Table 17 presents the topics considered for each assessment task: the associated risk, the risk-informed way of acting, the difficulties of assessment and the deficiencies of organisational support to assessing. The generic model is used as a reference in the identification of the most essential situation-specific features of management of fire situation in a particular fire scenario.

Table 17. The tabular presentation form in applying the reference model to a particular fire scenario.

Most relevant assessment tasks in the situation	Risk associated with the assessment task	Risk-informed way of carrying out the assessment task	Difficulties related to the assessment task	Deficiencies of organisational support
Tasks related to management of fire and to process control, selected and prioritised according to the scenario-specific conditions	Consequences, from the risk point of view, due to mis-judgements, delayed or omitted assessments	The way of acting and of the related co-operating with the fire fighting organisation, based on the recognition of the potential scenario-specific risks	Missing and unreliable information, difficulty to foresee the impact of the fire on the plant process, etc.	Deficiencies of procedures, training, information support systems and division of work

Risk-informed categorisation of systems, structures and components

Risk-informed categorization is based on utilising PSA information in a consistent way to select most cost-effective methods to control risk associated with systems, structures and components [9]. It provides a complementary method to the safety classification used in nuclear power plants. The approach developed is presented in a separate article of this SAFIR interim report.

Reliability of computer-based systems

Computer-based systems have an increasing influence to the operation of nuclear power plants. With programmable technology it is fairly straightforward to implement the functionality required for controlling different processes of a plant. On the other hand, at the same time programmable technology is an easy way of introducing unwanted complexity and unreliability to the instrumentation and control systems. For several years a reliability estimation methodology based on Bayesian inference has been developed at VTT [10–11]. In PPRISMA project the methodology is applied from PSA point of view [12]. With the methodology justified quantitative reliability estimates for safety functions implemented on programmable technology will become available.

The reliability estimation method is applied and developed in a form of case studies. First case study was started in 2004 and results of the study will be published in the beginning of 2005. In the case study software reliability of a motor protection relay is estimated. The case study is an in-kind contribution of ABB Substation Automation and the expert judgements of the different developer groups of the relay are applied as the prior estimate of the failure probability of the relay. The prior estimate is built in a special expert judgement process where the technical documentation of the relay is reviewed and uncertainties within and between the development groups are recognized and estimated. The prior reliability estimate given by the experts is updated to posterior estimate with available reliability data from relay testing and operational experience.

Applications

The results of the PPRISMA project have direct applications in risk-informed decision making both at nuclear safety authorities and utilities. The approaches developed in analysis of maintenance history information can be used for analysis, learning and prevention of multiple error mechanisms and enhancement of defensive barriers against the human common cause failures. The BIR study demonstrated a joint analysis of data on risk importance from PSA and on maintenance activities from the plant information system for equipment in component cooling water system.

The method developed to support risk-informed management of fire situations can be utilised in the development of the operators' training and procedures, the fire alarm system and the division of labor in the main control room. In addition, it supports the development of co-operation between the control room operators and the fire fighting organisation and the improvement of the realism of HRA for fire PSA.

The approach developed to the risk-informed categorisation complies with the Finnish regulatory PSA guide, which asks for an assessment of safety classification using PSA. In general, the risk-informed categorisation should be seen as an approach to support selection of appropriate and cost-effective methods to control risk.

The reliability estimation method for computer-based systems provides justified failure probabilities to be used in quantitative reliability analysis of safety functions implemented using programmable technology. Same time, the method provides guidance on the evidence needed for a good reliability estimate.

Conclusions

The challenges of risk-informed decision making can be discussed in two levels: 1) decision making practices when using risk-information and 2) quality of risk-information. Formal decision analysis methods developed in the project is one prerequisite for successful applications. Another important factor is the promotion of multi-disciplinary expert co-operation. Jointly developed reference models can facilitate the co-operation and communication. The quality of risk-information can be improved by collection of operating experience, by applying appropriate probability models and by focusing on informative and pedagogic presentation of results. The Bayesian framework could be utilised more e.g. in software reliability and human reliability assessment, as demonstrated in the project.

The PPRISMA project has been also active in international co-operation through e.g. OECD/NEA Working Groups Risk (WGRISK) and Operating Experience (WGOE), NKS, Nordic PSA Group (NPSAG), European Union Joint Research Centre (EU-JRC) in Petten and European Safety, Reliability and Data Association (ESReDA). Results have been presented internationally at the most important conferences in this field.

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25.2 Risk-informed categorisation

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Abstract

Risk-informed categorisation is based on utilising PSA information in a consistent way to select most cost-effective methods to control risk associated with systems, structures and components. In the presented method, an item's risk significance is broken in two parts: probability of failure and the consequence of failure. Higher value in either will possibly mean a higher risk category, depending on the limit values for each category. Risk categorisation aims at identification of items for improved reliability or added redundancy, versus the mostly reliability enhancing measures in safety classification.

Introduction

In many countries, regulatory decision making processes are being revised in response to the developments occurring in the nuclear energy field, where a restructuring is in progress due to market deregulation and plans of extending operating licenses and plant lifetimes. Following the developments by the United States Nuclear Regulatory Commission [1, 2], many countries are embarking on the implementation of *risk-informed regulation*. By this, regulatory bodies expect to increase the effectiveness of regulation. The objective of the risk-informed regulation is to define requirements that are consistent with the risk importance of the equipment, events and procedures to which the requirements will be applied.

A particular risk-informed application is to assess the safety classification of components, systems and structures and related QA requirements. The operating nuclear power plants have experience that their safety classification system and QA requirements are partly unbalanced and inflexible. This has been demonstrated also using PSA [3, 4]. However, to propose changes in the safety classes using risk information is difficult for an operating plant. On the other hand, risk-informed evaluations should be seen as a piece of information to consider possible other changes related to safety classes and the QA-system, e.g., optimisation of test intervals, operational rules, maintenance practices and in-service inspection strategies. For a new plant, the situation is different as long as the details of the design will be fixed. Risk-informed evaluation provides a method to compare different design solutions including alternative allocations of safety class, QA and reliability requirements.

Which risk-information should be "processed" from PSA, how the process risk-information should be interpreted and how the risk-interpretation should be implemented in the decision making are relevant research topics. In particular, this paper discusses the meaning of risk-informed decision making, meaning of risk information and meaning of risk-informed classification. It should be noted that we distinguish between *safety classification* and *risk-informed categorisation*. The former is associated with the regulatory guide YVL 2.1 [5]. The latter is associated with the way risk importance measures can be used to categorise items. The linkage between the safety classification and risk-informed categorisation is also discussed.

Risk-informed decision making

It is possible to identify three levels in dealing of uncertainty in safety related decision making [6, 7]: 1) utility theory based optimisation, 2) multi-attribute value function optimisation, and 3) non-formalised risk-informed decision making. The utility theory approach is the most formal way. PSA-model yields the probability distribution of consequences of a decision option. The decision maker expresses preferences and risk attitudes in the form of a utility function, and the decision option with maximum expected utility is selected. All uncertainties are expressed in probabilities, and there is only one decision criterion: the expected utility. However, to define a utility function is a difficult exercise, and therefore this approach is seldom applied.

In the multi-attribute value function optimisation, core damage risk is one of the decision criteria that is weighted against other criteria, e.g., cost. The decision option with maximum value is selected. In this approach, the deterministic and probabilistic criteria can be imbedded into the value model, and it is feasible to make trade-off between them. It is possible to interpret e.g. the ALARA-type (As-Low-As Reasonable Achieved) [8] and *formal* risk-informed decision making approaches from this perspective. As with the utility function approach, the decision maker may be sometimes unwilling to specify trade-offs. Therefore an *informal* approach is most commonly used in risk-informed applications.

The third approach admits that the decision context is extremely complex and only part or some aspects of it can be described with exact models. The PSA-model yields the probabilities of consequences, but the validity and uncertainties of it are evaluated in a qualitative, case dependent way. The values and preferences of the decision maker are expressed informally. The decision is made by using case dependent decision criteria and decision panels or other group decision approaches are utilised (Figure 120). This approach represents rather well the *informal* risk-informed decision making. The risk categorisation approach outlined in this paper can be considered such an approach.

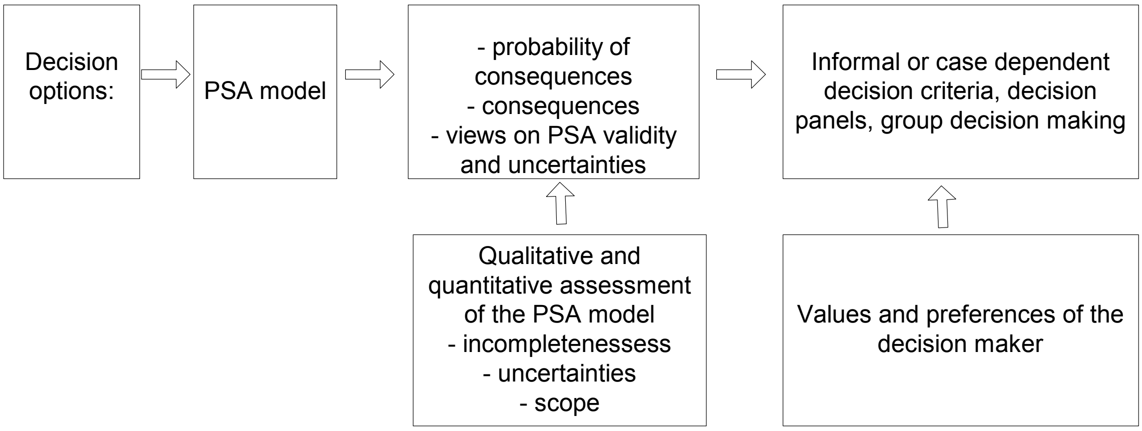

Figure 120. An informal approach to risk-informed decision making.

Risk information provided by PSA

The PSA model represents how combinations of failure events related to the safety critical functions, structures providing safety barriers and human interventions lead to an unwanted consequence. The model consists of a number of elementary items, called basic events in fault tree modelling. A top event, denoted here by TOP, represents the unwanted consequence, e.g., a core damage. The elementary items representing failure events potentially leading to the top event are denoted by $X_1, \ldots, X_n$. The probability of the top event is expressed by a probability function of the elementary items

$$P(TOP=1)=\sum P(TOP=1\mid X_1,\ldots,X_n)\cdot P(X_1,\ldots,X_n), \qquad (1)$$

where the summation on the right-hand side is taken over all possible combinations values of $X_1, \ldots, X_n$.

The equation (1) can decomposed with respect to any item, X_i, in the following way

$$P(TOP=1)=P(TOP=1\mid X_i=1)P(X_i=1)+P(TOP=1\mid X_i=0)P(X_i=0), \qquad (2)$$

assuming that X_i is a binary variable, as is the case when considering the basic events. The decomposition above is the key for analysing the risk importance of different items in the model. The terms in the equation (2) have the following interpretations:

- $P(X_i=1)$ is the probability of the failure event X_i, for instance, the unavailability of a component. $P(X_i=0)$ is the probability of the complementary event.
- $P(TOP=1\mid X_i=1)$ is the conditional probability of the top event given the failure event X_i. In PSA context, this measure is called safety margin, since it expresses

how much "safety margin" is left if the event takes place. It should be noted that the smaller the number is, the larger the safety margin is. Further, it should be noted that $P(TOP = 1 \mid X_i = 1)$ is proportional to the so called *risk increase factor* or *risk achievement worth* usually calculated for the basic events.

- $P(TOP = 1 \mid X_i = 0)$ is the conditional probability of the top event given the failure event X_i does not occur. This is also an interesting measure since it is related to the potential to reduce risk. In PSA, the corresponding risk importance measure is called *risk decrease factor*.

When analysing the results from PSA, the above quantities or quantities derived from the above conditional probabilities are used e.g. to rank the components from risk point of view. Consequently, the risk importance measures are applicable for categorising items. The question is which importance measures are chosen, which numerical criteria are chosen for the categories, and how risk categories are then used.

Time window for calculating probabilities

When using conditional core damage probability and component failure probability for risk importance measures, a time frame of interest must also be considered. Some intuitive candidates for a time window would be 1) the plant's lifetime (40–50 years), 2) a single operating year or 3) some arbitrarily small time. The two importance measures used in this paper are affected by the length of the time window in different ways, depending on the type of the component (one that causes an initiating event vs. one that is on stand-by to mitigate initiating events).

Failure probabilities ($P(X_i = 1)$) are evaluated in PSA either with existing reliability statistics, or with expert judgement if no data is available. For components that are modelled as basic events this figure is readily available. Components that cause an initiating event a failure frequency is used instead. Failure probability can be approximated from this by multiplying the frequency with the length of the time window of interest, $P(IE_i = 1) \approx f(IE) * T$. The time window of interest thus affects the failure probability of those components that cause an initiating event, while it does not affect the probability of components on stand-by.

Concerning the conditional core damage frequency ($P(TOP = 1 \mid X_i = 1)$), the situation is vice versa. For components that do not cause an initiating event a longer time window increases the conditional core damage probability. For components that lead into an initiating event the conditional core damage probability remains unaffected.

General concept for risk categorisation

A general concept for risk classification can be based on two quantities: a) the probability of the failure event X_i, $P(X_i = 1)$, and b) the conditional probability of the top event given the failure event X_i, $P(TOP = 1 \mid X_i = 1)$.

Figure 121 illustrates the alternative cases in an XY-diagram. The figure can be divided into four sections with the following interpretations:

A. Low probability, low consequences. Components relating to the basic event have a low contribution to the overall risk.

B. High probability, low consequences. Components in this category are not very reliable but the consequences of the failure are minor. Moderate overall risk.

C. Low probability, high consequences. Components in this class are reliable but their failures have serious consequences to the plants safety. Moderate overall risk.

D. High probability, high consequences. Components in this class have both high unreliability and high consequences for failure. High overall risk.

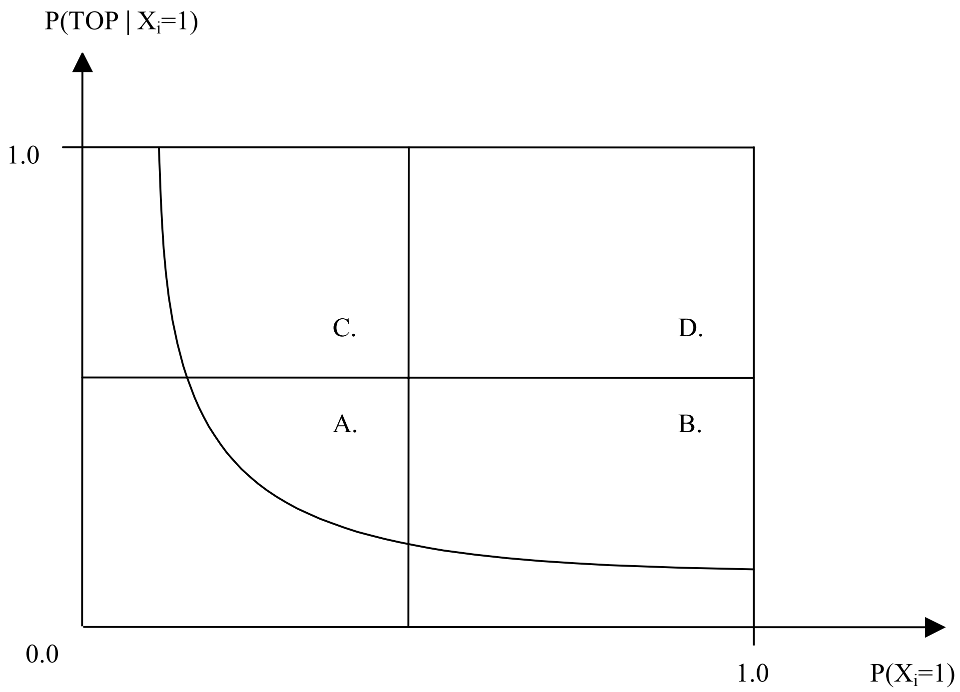

Figure 121. P(TOP = 1 | X_i = 1) vs. P(X_i = 1) plot. The four areas (A, B, C, D) corresponds to different risk categories. The curve is the uniform risk curve where the product $P(TOP = 1 \mid X_i = 1) \cdot P(X_i = 1)$ is constant.

The curve in Figure 121 is the uniform risk curve where the product $P(TOP = 1 \mid X_i = 1) \cdot P(X_i = 1)$ is constant. Each component on this curve has the same contribution to the overall risk of the system, e.g. they can be said to have the same risk importance. In fact, the Fussel-Vesely importance measure can be used as a measure for this risk, as it is constant along the curve.

Risk categorisation is used to control the overall risk in a systematic way. Each component that contributes to the risks in the system is classified based on its attributes, which are here the basic event probability and consequences. Other risk measures could also be used depending on which actions are available to control the risk. Each class has measures that reduce the risks associated with the component. In the risk categorisation method presented here the risk control methods would have to either reduce the probability of failure or the consequences of the failure. Measures that reduce risks (redundant components, components with higher quality) consume resources, so it is very importance to identify the targets where the resources are most needed.

Comparison of safety classification and risk categorisation

The purpose of existing safety classification is to ensure that the components, systems and functions have to be manufactured, installed and operated according to their safety significance. The four classes have increasing quality requirements for the components deemed to have greater safety significance. Thus, components with greater safety significance have a higher quality.

The regulatory guide YVL guide 2.1 does not explicitly state the risk significance, the product $P(TOP = 1 \mid X_i = 1)P(X_i = 1)$, of each component to be a factor in classification. However, the principle of reducing this risk is implied in the guide because components closer to the reactor core are assumed to have a greater safety significance, and are assigned to a higher safety class. Greater safety significance can be understood to mean a higher conditional core damage probability $P(TOP = 1 \mid X_i = 1)$. Measures associated with each safety class are mostly quality enhancing, which in terms of risk control affect the basic event probability $P(X_i = 1)$. So, the existing safety classification controls the risk associated with each component by affecting its reliability.

In risk informed categorisation candidates for two risk control methods can be identified. Both the probability of component failure and the consequence of component failure are considered. The four categories in Figure 121 can be used to decide which risk controls methods are utilized.

A. Low probability, low consequences. No risk reduction needed.

B. High probability, low consequences. Component reliability should be increased.

C. Low probability, high consequences. Consequences of failure should be reduced, by increasing redundancy, for example.

D. High probability, high consequences. Both reliability and redundancy should be enhanced.

Risk classification can be more efficient in reducing risks than the deterministic safety classification, because PSA studies have shown that the risk significances of components and systems do not follow the simple assumption that components closer to reactor have greater safety significance. However, it must be understood that the two principles are complementing each other, not competing with each other.

Different categories in this classification can not be ranked as the safety classes of YVL guide 2.1 can be. Categories B and C can both have similar risk significances, but the methods that should be used to reduce risks are different.

The use of resources is always implied in reducing risk and identifying the most important contributors to risk. In rational resource optimisation the resources used on reducing a components risk should be proportional to the decrease in risk. Burden-to-importance ratio (BIR) can be used for efficient resource allocation [9].

Conclusions

Two different classification schemes were discussed in this paper. The existing safety classification, required by YVL guide 2.1, is analysed in terms of its effect on risk. A new risk based categorisation was presented. A component's, system's or function's risk significance is broken in two parts for this categorisation: probability of failure and the consequence of failure. Higher value in either will possibly mean a higher risk category, depending on the limit values for each category. Risk categorisation allows for identification of components for either improved reliability or added redundancy, versus the mostly reliability enhancing measures in safety classification.

Risk categorisation and safety classification both aim at making the nuclear power plant safer. The safety classification is based on a deterministic safety analysis, while the risk categorisation presented here is based on the insights provided by a PSA study. Both analysis methods are essential in nuclear power plant safety. The risk categorisation method presented here is a step toward utilising PSA information in a consistent way.

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Appendix A: Publications of the projects

Enhanced methods for reactor analysis (EMERALD):

Scientific publications

Daavittila, A., Hämäläinen, A. & Kyrki-Rajamäki, R. Effects of secondary circuit modeling on results of PWR MSLB benchmark calculations with new coupled code TRAB-3D/SMABRE. Nuclear Technology, Vol. 142, No. 2, pp. 116–123, May 2003.

Vanttola, T., Hämäläinen, A., Kliem, S., Kozmenkov, Y., Weiss, F.-P., Keresztúri, A., Hádek, J., Strmensky, C., Stefanova, S. & Kuchin, A. et al. Validation of coupled codes using VVER plant measurements. To be published in Nuclear Engineering and Design, 2004.

Conference papers

Anttila, M. & Ranta-aho, A. Burnup calculations for a 10x10 BWR assembly – comparison of CASMO-4, Monteburns 1.0 and MCB results. Studsvik Scandpower Users Group Meeting, April 19–21, 2004, Turku, Finland. 21 p.

Chen, Y., Wasastjerna, F., Fischer, U. & Simakov, S. Three-dimensional shielding calculations for the IFMIF neutron source using a coupled Monte Carlo/Deterministic computational scheme. 10th International Conference on Radiation Shielding (ICRS-10) / 13th Topical Meeting on Radiation Protection and Shielding (RPS-2004). May 9–14, 2004, Funchal, Madeira.

Daavittila, A. TRAB-3D/TRAB pressure transient calculation. NKS-R 3D BWR Transient Methodology Seminar. April 8, 2003, Espoo, Finland. NKS-89, NKS Secretariat, Roskilde, Denmark, 2003. 12 p.

Hämäläinen, A. 3D methodology in VVER transients with uncertainty analyses. NKS-R 3D BWR Transient Methodology Seminar. April 8, 2003, Espoo, Finland.

Hämäläinen, A. & Kyrki-Rajamäki, R. HEXTRAN-SMABRE calculation of the 6th AER benchmark, main steam line break in a VVER440 NPP. Proceedings of the 13th Symposium of the AER on VVER Reactor Physics and Safety, September 22–26, 2003, Dresden, Germany. KFKI/AEKI, Budapest, 2003. Pp. 445–458. ISBN 963-372-630-1

Hämäläinen, A., Vanttola, T., Weiss, F.-P., Mittag, S., Kliem, S., Kozmenkov, Y., Langenbuch, S., Keresztúri, A., Hádek, J., Strmensky, C., Stefanova, S., Kuchin, A., Hlbocky, P., Siko, D. & Danilin, S. Validation of coupled codes in VALCO WP1 using measured VVER data. Proceedings of the 13th Symposium of the AER on VVER Reactor Physics and Safety, September 22–26, 2003, Dresden, Germany. KFKI/AEKI, Budapest, 2003. Pp. 555–576. ISBN 963-372-630-1

Höglund, R. VTT Processes, Solala, M., Dahlbacka, K. & Latokartano, S., TVO, Shutdown margin in the case of a misplaced bundle in a BWR core. Studsvik Scandpower Users Group Meeting, April 19–21, 2004, Turku, Finland. 10 p.

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Laakso, K. & Saarelainen, P. Inhimillisperäiset yhteisviat kunnossapitotoiminnan yhteydessä. Kunnossapito-historian järjestelmällinen tutkinta ja analyysi Loviisan voimalaitokselle (Human originated common cause failures related to maintenance activities. A systematic analysis of maintenance history for Loviisa nuclear power plant). VTT Research Report BTU062-041220, Espoo 8.1.2004. 107 p + app. 26 p. (In Finnish)

Männistö, I. Study on classification tasks. Report BTUO62-031196, VTT Industrial Systems, Espoo. 36 p.

Rosqvist, T. & Laakso, K. The Burden to Importance Ratio – a demonstrative study on BIR application for resource allocation of condition based maintenance. VTT Research Report 062-031189. Espoo 10.08.2004. 19 p.

Simola, K. & Pulkkinen, U. Summary of on-going activities in risk-informed decision making. Report BTUO62-031205, VTT Industrial Systems, Espoo. 3 p + app. 12 p.

Simola, K. & Pulkkinen, U. Risk-informed decision making – a pre-study. Report NKS-93(2004), Nordisk kernesikkerhedsforskining NKS, Roskilde. 15 p + app. 12 p.

Others

Laakso, K. Kustannustehokkuutta ja käytettävyyttä kokemuspohjaisella kunnossapitoanalyysillä. Kunnossapito 4/2003, pp. 40–43. (In Finnish)

Laakso, K. Euromaintenance 11.5.–13.5.2004. Huomioita konferenssimatkalta. (Notes from a conference journey). Kunnossapito 5/2004. Pp. 65–67. (In Finnish)

Laakso, K., Simola, K. & Hänninen, S. Maintenance Analysis of Technical Systems. In Kunnossapito 5/2002. Maintenance Research in Finland. Pp. 49–51.

Myötyri, E. Measures for structural properties of systems. M.Sc. Thesis. Helsinki University of Technology – Department of engineering physics and mathematics, Espoo, 2003. 52 p. + app. 20 p.

SAFIR Administration and information (SAHA):

Research institute reports

Puska, E.K. SAFIR, The Finnish Research Programme on Nuclear Power Plant Safety, Annual Plan 2003. Espoo: VTT Processes Project Report PRO1/P7007/03. 26 p. + app. 102 p.

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

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

Others

Puska, E.K. SAFIR koostaa parasta kansallista alan asiantuntemusta. Energia. (2003) 4–5, pp. 35. (In Finnish)

Appendix B: International co-operation connections

OECD/Nuclear Energy Agency

Committee on Nuclear Regulatory Activities (CNRA), J. Laaksonen STUK (Chairman), L. Reiman, STUK

* Working Group on Inspection Practices, S. Suksi, STUK
* (CNRA/CSNI) Task Group on Safety Performance Indicators, S. Suksi, STUK
* Task Group on Regulatory Effectiveness Indicators, K. Koskinen, STUK

Committee on the Safety of Nuclear Installations (CSNI), L. Mattila, VTT Processes, K. Valtonen, STUK

* Working Group on Operating Experience, K. Laakso, VTT Industrial Systems, T. Eurasto, STUK, Task Group on IRS, K. Tossavainen, STUK, Task Force on Computer Based Safety Systems, M.-L. Järvinen, STUK
* Working Group on Risk Assessment, J.-E. Holmberg, VTT Industrial Systems, R. Virolainen, STUK
* Working Group on Analysis and Management of Accidents, M. Tuomainen, VTT Processes, N. Lahtinen, STUK
* Working Group on Integrity and Ageing of Components and Structures, R. Rintamaa, VTT Industrial Systems, R. Rantala, STUK

  Task Group on Integrity of Metal Components and Structures, J. Solin, VTT Industrial Systems, R. Rantala, STUK
* Special Expert Group on Human and Organisatorial Factors, L. Norros, VTT Industrial Systems, K. Åstrand, STUK
* Special Expert Group on Fuel Safety Margins, S. Kelppe, VTT Processes
* WG-1 International Working Group on ECCS Sumps and Strainers, J. Hyvärinen, STUK
* Writing Group of the CSNI State-of-the-Art Report on Nuclear Aerosols in Reactor Safety, J. Jokiniemi & A. Auvinen, VTT Processes

Committee on Radiation Protection and Public Health (CRPPH), O. Vilkamo, STUK

Nuclear Development Committee (NDC), J. Aurela, KTM, M. Anttila, VTT Processes

Nuclear Science Committee (NSC), M. Anttila, VTT Processes

* Working Party on Scientific Issues in Reactor Systems (WPRS), A. Daavittila, VTT Processes
* Working Party for Nuclear Criticality Safety, R. Mattila, STUK
* Task force on scientific issues of fuel behaviour, S. Kelppe, VTT Processes

Information System on Occupational Exposure (ISOE), V. Riihiluoma, STUK, K. Alm-Lytz, STUK

CSNI International Common-Cause Failure Data Exchange (ICDE) Project, K. Jänkälä, FORTUM Nuclar Services, J. Pesonen, TVO, R. Virolainen, STUK

Halden Reactor Project/Halden Programme Group, Olli Ventä, VTT Industrial Systems, T. Vanttola, VTT Processes

* *Advanced Monitoring techniques for Application in Material Studies,* K. Mäkelä, VTT Industrial Systems
* *Pressure Vessel Ageing,* M. Valo, VTT Industrial Systems
* *Irradiation Assisted Stress Corrosion Cracking,* P. Aaltonen, VTT Industrial Systems
* *Fuel performance analysis,* S. Kelppe, VTT Processes
* *Reliability of software based control systems,* A. Helminen, VTT Industrial Systems
* *Integrated system validation, Innovative Displays,* L. Norros & P. Savioja, VTT Industrial Systems

MASCA Project, Programme Review Group (Chairman), Management Board, H. Tuomisto, Fortum Nuclear Services

MCCI Project, Management Board, Programme Review Group, I. Lindholm, VTT Processes

SETH Project, Management Board, E. Virtanen, STUK, Programme Review Group, H. Purhonen, LUT

International Atomic Energy Agency

Nuclear Safety Standards Committee, L. Reiman, STUK

IAEA Extrabudgetary Program on Mitigation of Intergranular Stress Corrosion Cracking in RBMK Reactors, R. Keskinen, STUK

International Nuclear Event Scale (INES), K. Tossavainen, STUK

Incident Reporting System (IRS), K. Tossavainen, STUK

Incident Reporting System for Research Reactors (IRSRR), K. Tossavainen, STUK

Technical Working Group on Nuclear Power Plant Control and Instrumentation (TWG-NPPCI), B. Wahlström (Chairman), VTT Industrial Systems

Technical Working Group on Advanced Light Water Reactors (TWG-LWR), E. Patrakka (Chairman), TVO

Co-ordinated Research Projects
* *Round-robin Exercise on WWER-440 RPV Weld Metal Irradiation Embrittlement, Annealing, and Re-embrittlement*, M. Valo, VTT Manufacturing Technology
* *International Working Group of Life Management of Nuclear Power Plants (IWG-LMNPP),* K. Wallin, VTT Industrial Systems
* *High temperature On-line Monitoring of Water Chemistry and Corrosion, (WACOL),* K. Mäkelä, VTT Industrial Systems
* *CRP Coordinated Research Programme "Assuring Structural Integrity of Reactor Pressure Vessels",* K. Wallin, VTT Industrial Systems
* *CRP Coordinated Research Programme "Scientific Basis and Engineering Solutions for Cost-effective Assessments of Software Based I&C Systems",* H. Harju, VTT Industrial Systems
* *CRP FUMEX II; Coordinated Research Programme on "Improvement of models used for fuel behaviour simulation",* S. Kelppe, VTT Processes
* *International Working Group on Water Reactor Performance and Technology (IWGFPT),* R. Teräsvirta, Fortum Engineering, S. Kelppe, VTT Processes

FARO Expert Group, T. Karjunen, STUK

Commission of the European Communities

DG Enegy and Transport, European Forums

* *European Plant Life and Ageing Forum (EPLAF),* R. Rintamaa, VTT Industrial Systems
* *European NDE Forum (ENDEF),* P. Kauppinen, VTT Industrial Systems

DG Environment
* *Working group codes and standards (WGCS)*, R. Rintamaa, VTT Industrial Systems

DG Research
* *JRC Steering committee*, J. Forstén, VTT
* *Networks coordinated by JRC/IAM*, R. Rintamaa, VTT Industrial Systems
* *Network for Evaluating Steel Components (NESC),* AG1 Inspection, P. Kauppinen & J. Pitkänen, VTT Industrial Systems
* *European Network for Inspection Qualification (ENIQ) Steering committee,* K. Hukkanen, Teollisuuden Voima Oy
* *European Network for Inspection Qualification (ENIQ) TGR, ENIQ task group for Risks*, M. Sarkimo & K. Simola, VTT Industrial Systems
* *Effective application of TOFD method for weld inspection at the manufacturing of pressure vessels,* P. Kauppinen, VTT Industrial Systems

Nuclear Fission Safety in the Sixth Framework Programme

* *Consultative Committee Euratom-Fission (CCE-Fission)*, J. Aurela & A. Väätäinen, KTM

Nuclear Regulators Working Group (NRWG), P. Koutaniemi, STUK

* *Task Force on NDT Qualification Programmes,* O. Valkeajärvi, STUK
* *Task Force on Safety Critical Software – Licensing Issues,* M.-L. Järvinen, STUK

Group of Experts under Article 31 of the Euratom Treaty, O. Vilkamo, STUK

JRC-Ispra-ISID, Reactor Safety Programme Users Advisory Board (RSPUAB), R. Virolainen, STUK

Phebus FP Project, J. Jokiniemi & A. Auvinen, VTT Processes

* Scientific analysis working group (SAWG)
* Bundle interpretation circle (BIC)
* Circuit and containment interpretation circle (CACIC)
* Containment chemistry interpretation circle (CCIC)
* Air ingress working group (AIWG)
* Air ingress task force (AITF)

Cooperation on VVER Reactor Physics and Dynamics (AER)

Scientific council, H. Räty, VTT Processes, P. Siltanen, Fortum NS

Nordic Nuclear Safety Research (NKS)

Steering group, U. Ehrnstén, VTT Industrial Systems, J. Aurela, KTM, O. Vilkamo, STUK

Research Programme on Reactor Safety, 2002 – , P. Lundström (Programme manager), Fortum Nuclear Services

TUD (Informationsystem för Tillförlitlighet, Underhåll och Drift), J. Pesonen, TVO

NPSAG (Nordiska PSA gruppen), R. Himanen & J. Pesonen, TVO

Experiements on Ruthenium behaviour, U. Backman, J. Jokiniemi, A. Auvinen, & R. Zilliacus, VTT Processes

Scientific Communities

European Safety, Reliability and Data Association (ESReDA)

*	*Executive Committee, Organisation of yearly ESReDA Seminars,* K.Simola, VTT Industrial Systems

Probabilistic safety assessment and management (PSAM) and Probabilistic safety assessment (PSA) conferences

*	*Organising committ*ee, Reino Virolainen, STUK

Technical Program Committee on PSAM6 & ESREL 2003 Conferences, Member, J. Vaurio, Fortum Power and Heat

European Structural Integrity Society (ESIS), K. Wallin, H. Talja, VTT Industrial Systems

*	*Development of European "standards" on fracture mechanics, information exhance.* VTT co-chairs the Materials Task Group and articipates in the Numerical task Group

International Group for Radiation Damage Mechanisms in Pressure Vessel Steels (IGRDM), K. Wallin & M. Valo, VTT Industrial Systems

ASTM, K. Wallin, E-10, M. Valo, VTT Industrial Systems

*	*subcommittee E-10 on Nuclear Technology* concentrates on monitoring of irradiation embrittlement using small specimens and developes related standards

ASME, R. Rintamaa, VTT Industrial Systems

*	*participation in yearly PVP (Pressure Vessels and Piping) conferenses*

Cooperation with various institutes

SCK

*	CEN, Studiecentrum voor Kernenergie, Belgium
*	*Development of advanced monitoring techniques*, K. Mäkelä, VTT Industrial Systems

Paul Sherrer Institut, Switzerland

*	*European Round Robin on Constant Load EAC Tests of Low Alloy Steel under BWR Conditions*, P. Karjalainen-Roikonen & U. Ehrnstén, VTT Industrial Systems

Leningrad NPP Sosnovyj Bor, Russia

*	*The Finnish-Russian co-operation on integrity of pressurised components*, P. Kauppinen, VTT Manufacturing

Institute de Radioprotection et de Sûreté Nucléaire, Cadarache, France

- *Design and testing of ruthenium measuring instrumentation*, A. Auvinen, J. Jokiniemi & U. Backman, VTT Processes.
- *Study of the behaviour of highly irradiated fuels in case of reactivity accident and the SCANAIR computer code*, S. Kelppe, VTT Processes
- *OECD-IPSN CABRI Water Loop Project 2000–2007. Umbrella Agreement with OECD, bilateral Agreement with IPSN; jointly with Fortum Power and Heat Oy and Teollisuuden Voima Oy*. K. Valtonen, STUK (Steering Committee), S. Kelppe, VTT Processes (Technical Advisory Group)

Research Institute of Technology, NITI, Russia

- *Scientific cooperation on thermal-hydraulic experiments,* H. Purhonen, LTKK

US Nuclear Regulatory Commission (USNRC)

- PIRT Panel (on fuel burnup), K. Valtonen, STUK
- *Code Application and Maintenance (CAMP),* H. Holmström, VTT Processes
- *Co-operative Severe Accident Research Programme (CSARP),* I. Lindholm, VTT Processes
- *FRAPCON-3/FRAPTRAN Code Users' Group*, S. Kelppe, VTT Processes
- *FRAPTRAN/GENFLO Fuel Performance CodeDevelopment,* S. Kelppe, VTT Processes
- *International Collaborative Project to Evaluate Fire Models for Nuclear Power Plant Applications organized by USNRC,* O. Keski-Rahkonen, VTT Building Technology

Electric Power Research Institute (EPRI)

- *Advanced Containment Experiments, Extension (ACEX),* I. Lindholm, VTT Processes
- *Melt Attack and Coolability (MACE),* I. Lindholm, VTT Processes
- *Cooperative Irradiation Assisted Stress Corrosion Cracking (IASCC) Research Programme (CIR),* P. Aaltonen, VTT Industrial Systems

Swedish Nuclear Power Inspectorate (SKI), Sydkraft, OKG and Vattenfall Ab, Sweden

- *SKI Forskningnämnd*, L. Norros, VTT Industrial Systems
- *SKI Reaktorsäkerhetsnämd*, L. Reiman, STUK

Staatliche Materialprüfungsanstalt (MPA), Germany

- *Materials research,* R. Rintamaa, VTT Industrial Systems

Fraunhofer-Institut für Werkstoffmechanik (IWM), Germany

- *Structural analysis and computational fracture mechanics, especially development of new material models*, H. Talja, VTT Industrial Systems

University of Illinois, USA

- C*omputational fracture mechanics, assessment of damage,* K. Wallin, VTT Industrial Systems

Forschungszentrum Karlsruhe (FZK), Germany
* *Hydrogen detonation simulation*, A. Silde, VTT Processes

VGB SWR-Arbeitskreis, Germany
* A. Reinvall, TVO

ITU, Karlsruhe, A. Auvinen & J. Jokiniemi, VTT Processes.
* Revaporisation of fission products from Phebus FP samples

National Institute of Standards and Technology (NIST), USA
* *Development of Fire Dynamics Simulator*
* *Direct numerical simulation of flame spread on cylindrical wood rods,* S. Hostikka, VTT Building and Transport

ALARA Engineering and Advanced Nuclear Technology, Sweden, M. Bojinov, VTT Industrial Systems
* *Development of a predictive model of activity incorporation and corrosion phenomena*

IVF – Industriforskning och utveckling AB
* *No Lead in Nordic Electronics*, A. Turtola, VTT Industrial Systems.

SINTEF – Stiftelsen for industriell og teknisk forskning ved Norges tekniske høgskole
* *No Lead in Nordic Electronics*, A. Turtola, VTT Industrial Systems.

DELTA – Danish Electronics, Light and Acoustics
* *No Lead in Nordic Electronics*, A. Turtola, VTT Industrial Systems.

Other co-operation

European Nuclear Installations Safety Group (ENIS-G), P. Koutaniemi, STUK

EC/TC45/SC45A/Working Group A3, H. Heimbürger, STUK

IEC/TC45/SC45A/Working Group A10, H. Palmén, VTT Industrial Systems.

IEC/TC45/SC45A, Nuclear Instrumentation Committee (SESKO), M.-L. Järvinen, STUK

Technical Safety Organisation Group (TSOG), Technical Committee, H. Saari, STUK

European Working Group on Reactor Dosimetry – Programme Committee (EWGRD-PC), T. Serén, VTT Processes

Working Group on Reactor Dosimetry for VVER Reactors (WGRD-VVER), T. Serén, VTT Processes

European Network of Testing Facilities for the Quality Checking of Radioactive Waste Packages (ENTRAP), A. Tiitta, VTT Processes

International Co-operative Group on Environmentally Assisted Cracking of Light Water Reactor Materials (ICG-EAC), P. Aaltonen, VTT Industrial Systems

International Group on Radiation Damage Mechanisms in Pressure Vessel Steels (IGRDM), M. Valo & K. Wallin, VTT Industrial Systems

Nordic Reactor Physics Meetings "Reactor Physics Calculations in the Nordic Countries", R. Höglund, VTT Processes

European Association of Cognitive Ergonomics (EACE), L. Norros, VTT Industrial Systems

New Technology and Work (NeTWork), L. Norros, VTT Industrial Systems

Nordic ALEX-group on advanced alara-princip in chemistry and radiation (Westinghouse Atom, Alara-Engineering, WA-BWR-plants), A. Reinvall, TVO

The human factors network for the process industries (PRISM), co-ordinated by the European Process Safety Centre (EPSC), K. Ruuhilehto, VTT Industrial Systems.

BWR OG PSA (BWR Owners Group, PSA task), R. Himanen, TVO

ISTC Project #833 METCOR "Investigation of corium melt interaction with NPP reactor vessel steel, NITI, Sosnovy Bor, Russia, Collaborator and Steering Committee Member, H. Tuomisto, Fortum Nuclear Services

MSWI (Melt-Structure-Water Interaction) Project, KTH, Stockholm, Advisory Group, H. Tuomisto, Fortum Nuclear Services

Appendix C: Academic degrees awarded in the projects 1.1.2003–8.12.2004

Enhanced methods for reactor analysis (EMERALD)

Licentiate in Technology:

Leppänen, J. Systematic comparison of evaluated nuclear data files. Helsinki University of Technology, Licentiate's Thesis, 28 May 2004. (VTT Project Report PRO1/P7009/04). 104 p.

Master of Science in Technology:

Ranta-aho, A. Validation of reactor physics codes for predictions of isotopic compositions of high burnup LWR fuels. Helsinki University of Technology, Master's Thesis, 1 February 2004. (VTT Project Report PRO1/P7006/04). 147 p.

Integrity and lifetime of reactor circuits (INTELI)

Doctor of Technology:

Moilanen, P. Pneumatic servo-controlled material testing device capable of operating at high temperature water and irradiation conditions. Doctoral thesis. VTT Publications 532, Espoo 2004.

Toivonen, A. Stress corrosion crack growth rate measurement in high temperature water using small preracked bend specimens. Doctoral thesis. VTT Publications 531, Espoo 2004.

Development of Aerosol Models to Nuclear Applications (AMY)

Master of Science in Technology:

Siltanen, S. Influence of Uncertainty of Calculation Parameters in Modelling on Radioactive Releases in Severe Accident Conditions. Helsinki University of Technology, Master's Thesis, 2 December 2003. 93 p. + app. (In Finnish)

Condensation pool experiments (POOLEX)

Master of Science in Technology:

Nurminen, T. Passive Safety Features of VVER-640 Reactors (In Finnish). Lappeenranta, 2003. Lappeenranta University of Technology. 80 p.

Räsänen, A. Measurement System for Steam Condensation (In Finnish). Lappeenranta, 2004. Lappeenranta University of Technology. 83 p. + app 2 p.

Interaction approach to development of control rooms (IDEC)

Master of Science in Technology:

Savioja, P. Käyttäjäkeskeiset menetelmät monimutkaistenjärjestelmien vaatimusten kuvaamisessa. Espoo, 2003. Helsinki University of Technology. 119 p.

Potential of fire spread (POTFIS)

Doctor of Philosophy:

Mangs, J. 2004. On the fire dynamics of vehicles and electrical equipment. VTT Publications 521, VTT Building and Transport, Espoo. (University of Helsinki). 62 p. + app. 101 p.

Principles and Practices of Risk-Informed Safety MAnagement (PPRISMA)

Doctor of Technology:

Rosqvist T. On the use of expert judgement in the qualification of risk assessment. Espoo: Technical Research Centre of Finland, 2003. VTT Publications 507. (Helsinki University of Technology). 48 p + app. 82 p. ISBN 951-38-6243-7,951-38-6244-5

Master of Science in Technology:

Myötyri, E. Measures for structural properties of systems. Espoo, 2003. Helsinki University of Technology – Department of engineering physics and mathematics. 52 p. + app. 20 p.

Appendix D: The Steering group, the reference groups and the scientific staff of the projects

Steering Group of SAFIR – The Finnish Research Programme on Nuclear Power Plant Safety 2003–2006

Kansallisen ydinvoimalaitosten turvallisuustutkimusohjelman SAFIR 2003–2006 johtoryhmä

Person	Organisation & Finnish abbreviation
Lasse Reiman, Chairperson	Radiation and Nuclear Safety Authority of Finland (STUK)
Timo Okkonen, Chairperson	*Radiation and Nuclear Safety Authority of Finland (STUK)**
Marja-Leena Järvinen	Radiation and Nuclear Safety Authority of Finland (STUK)
Piia Moilanen	National Technology Agency of Finland (TEKES)
Reijo Munther	National Technology Agency of Finland (TEKES)
Timo Vanttola	Technical Research Centre of Finland (VTT)
Heli Talja	Technical Research Centre of Finland (VTT)
Eero Patrakka	Teollisuuden Voima Oy (TVO)
Marjo Mustonen	Teollisuuden Voima Oy (TVO)
Harriet Kallio	Fortum Power and Heat Oy (Fortum P&H)
Jyrki Kohopää	Fortum Nuclear Services Oy (Fortum NS)
Rainer Salomaa	Helsinki University of Technology (TKK)
Riitta Kyrki-Rajamäki, Vice Chairperson	Lappeenranta University of Technology (LTY)
Anne Väätäinen	Finnish Ministry of Trade and Industry (KTM)
Jorma Aurela, KTM contact person	Finnish Ministry of Trade and Industry (KTM)

** resigned from STUK during 2003*

SAFIR Reference Groups
SAFIR tukiryhmät

1. Reactor fuel and core
Polttoaine ja reaktorisydän

Keijo Valtonen	STUK	Chairperson
Pertti Siltanen	Fortum	Vice Chairperson
Nina Lahtinen	STUK	
Risto Teräsvirta	Fortum	
Martti Antila	Fortum	
Kari Ranta-Puska	TVO	
Mikael Solala	TVO	
Seppo Koski	TVO	
Jaakko Miettinen	VTT PRO	
Lena Hansson-Lyyra	VTT TUO	

2. Reactor circuit and structural safety
Reaktoripiiri ja rakenteellinen turvallisuus

Martti Vilpas	STUK	Chairperson
Rainer Rantala	STUK	Vice Chairperson
Kirsti Tossavainen	STUK	
Erkki Kaminen	Fortum	
Alpo Neuvonen	Fortum	
Ossi Hietanen	Fortum	
Juho Hakala	TVO	
Erkki Muttilainen	TVO	
Timo Pättikangas	VTT PRO	
Rauno Rintamaa	VTT TUO	
Kim Wallin	VTT TUO	
Hannu Hänninen	TKK	
Jukka Tuhkuri	TKK	

3. Containment and process safety functions
Suojarakennus ja prosessiturvatoiminnot

Olli Nevander	TVO	Chairperson
Juhani Hyvärinen	STUK	Vice Chairperson
Nina Lahtinen	STUK	
Hannu Ollikkala	STUK	
Lauri Pöllänen	STUK	
Heikki Sjövall	TVO	*Chairperson in 2003*

Petra Lundström	Fortum	
Olli Kymäläinen	Fortum	
Antti Daavittila	VTT PRO	
Hanna Räty	VTT PRO	
Olavi Keski-Rahkonen	VTT RTE	
Juhani Vihavainen	LTY	
Pekka H. Pankakoski	VTT TUO	

4. Automation, control room and information technology
Automaatio, valvomo ja tietotekniikka

Esko Rinttilä	Fortum	Chairperson
Olli Hoikkala	TVO	Vice Chairperson
Harri Heimbürger	STUK	
Erik Lönnqvist	STUK	
Heimo Takala	STUK	
Jukka Kupila	STUK	
Martti Välisuo	Fortum	
Juha Miikkulainen	TVO	
Sixten Norrman	VTT PRO	
Risto Sairanen	*VTT PRO**	
Olli Tiihonen	VTT PRO	
Olavi Keski-Rahkonen	VTT RTE	
Jan-Erik Holmberg	VTT TUO	

** resigned from VTT during 2003*

5. Organisations and safety management
Organisaatiot ja turvallisuuden hallinta

Matti Vartiainen	TKK	Chairperson
Anneli Leppänen	TTL	Vice Chairperson
Timo Eurasto	STUK	
Kaisa Åstrand	STUK	
Jari Snellman	Fortum	
Pekka Luukkanen	Fortum	
Markku Friberg	TVO	
Urho Pulkkinen	VTT TUO	
Leena Norros	VTT TUO	

6. Risk-informed safety management
Riskitietoinen turvallisuuden hallinta

Reino Virolainen	STUK	Chairperson
Ilkka Niemelä	STUK	Vice Chairperson
Jouko Marttila	STUK	
Jussi Vaurio	Fortum	
Kalle Jänkälä	Fortum	
Risto Himanen	TVO	
Kari Taivainen	TVO	
Ilona Lindholm	VTT PRO	
Arja Saarenheimo	VTT TUO	
Olli Ventä	VTT TUO	
Esko Mikkola	VTT RTE	

Personnel and tasks in the SAFIR projects in 2003–2004

Enhanced methods for reactor analysis (EMERALD)
Kehittyneet reaktorianalyysimenetelmät

Research organisation: VTT Processes
Project manager: Randolph Höglund, VTT Processes
Deputy project manager: Antti Daavittila, VTT Processes

Person	Org.	Task
Randolph Höglund, LicTech	VTT	Project manager, reactor physics, nodal methods, Nordic connections
Antti Daavittila, MScTech	VTT	Deputy project manager, reactor dynamics, development and validation of dynamics codes
Markku Anttila, MScTech	VTT	Reactor physics, cross sections, isotope concentrations, OECD/NEA connections: NDC, NSC
Anitta Hämäläinen, MScTech	VTT	Reactor dynamics, circuit modelling, thermal hydraulic modelling for fuel transient codes, dynamics benchmarks
Elja Kaloinen, MScTech	VTT	Reactor physics, nodal methods, reactor physics in dynamics codes
Petri Kotiluoto, MSc (2004)	VTT	Reactor physics, transport methods, development of MultiTrans
Jaakko Leppänen, LicTech	VTT	Reactor physics, cross sections, use of Monte Carlo methods in burnup calculations
Riku Mattila, MScTech (2003)	VTT	Reactor physics, advanced nodal methods (AFEN, ARES)
Jaakko Miettinen, MScTech	VTT	Reactor dynamics, development of the coupled TRAB-3D/SMABRE code

Markku Rajamäki, DTech	VTT	Reactor dynamics, development, testing and application of CFDPLIM
Anssu Ranta-aho, MScTech	VTT	Reactor physics, criticality safety, isotope concentrations
Hanna Räty, MScTech	VTT	Reactor dynamics, development and validation of dynamics codes, AER connections
Elina Syrjälahti, MScTech	VTT	Reactor dynamics, thermal hydraulics modelling, dynamics benchmarks
Timo Vanttola, DTech	VTT	Reactor dynamics, special questions on thermal hydraulics
Frej Wasastjerna, LicTech	VTT	Reactor physics, MCNP (Monte Carlo calculations)

High Burnup Updates in Fuel Behaviour Modelling (KORU) 2003
Polttoaineen korkeapalamamallinnuksen uudistaminen

Research organisation: VTT Processes
Project manager: Seppo Kelppe, VTT Processes
Deputy project manager: Jan-Olof Stengård, VTT Processes

Person	Org.	Task
Seppo Kelppe, MScTech	VTT	Project Manager; ENIGMA devel.and appl.
Arttu, Knuutila, MScTech	VTT	Mechanical modelling, Materials; SCANAIR
Jan-Olof Stengård, assistant research scientist	VTT	FRAPTRAN development and applications
Laura Kekkonen, research trainee	VTT	Fuel steady-state modelling; ENIGMA
Jaakko Miettinen, LicTech	VTT	GENFLO development and applications
Maiju Seppälä, research trainee	VTT	Part-time trainee

Integrity and life time of reactor circuits (INTELI)
Reaktoripiirin eheys ja käyttöikä

Research organisations: VTT Industrial Systems and VTT Processes (VTT PRO)
Project manager: Pentti Kauppinen, VTT Industrial Systems
Deputy project manager: Heikki Keinänen, Arja Saarenheimo, Pertti Aaltonen VTT Industrial Systems

Person	Org.	Task
INSEL		
Heikki Keinänen, MScTech	VTT TUO	INSEL sub-project manager
Matti Valo, MScTech	VTT TUO	Research on ageing mechanisms
Kim Wallin, DTech	VTT TUO	Modeling of ageing

Anssi Laukkanen, MScTech	VTT TUO	Transfer of test results for structural analysis
Pekka Nevasmaa, DTech	VTT TUO	Applicability of small specimen test results
Tapio Planman, MScTech	VTT TUO	Ageing mechanisms
Pertti Aaltonen, MScTech	VTT TUO	Integrity of bimetal welds
Ulla Ehrnsten, MscTech	VTT TUO	Integrity of bimetal welds
Toni Hakkarainen, technician	VTT TUO	Ultrasonic TOFD-technique
Jorma Pitkänen, LicTech	VTT TUO	Ultrasonic analysis techniques
Tom Seren	VTT PRO	Dosimetry
Lena Hansson-Lyyra	VTT TUO	Halden research on fuel capsule corrosion
INPUT		
Arja Saarenheimo, LicTech	VTT TUO	INPUT sub-project manager
Matti Sarkimo, LicTech	VTT TUO	Ultrasonic simulation
Marko Api, technician	VTT TUO	Ultrasonic simulation
Kaisa Simola, DTech	VTT TUO	Risk informed approach to ISI
Otso Cronvall, MScTech	VTT TUO	Risk informed approach to ISI
Kim Calonius, MScTech	VTT TUO	Numerical analysis
Timo Pättikangas, MSc	VTT PRO	Fluid-structure interaction analysis
Petri Kinnunen, DTech	VTT TUO	Oxide film growth on stainless steel
Timo Laitinen, DTech	VTT TUO	Oxide film growth on stainless steel
Timo Saario, DTech	VTT TUO	Water chemistry – corrosion interaction
Kari Mäkelä, DTech	VTT TUO	Water chemistry – corrosion interaction
Pekka Moilanen, DTech	VTT TUO	Research on hydrodynamics
Martin Bojinov, DTech	VTT TUO	Oxide film growth on stainless steel
Aki Toivonen, DTech	VTT TUO	Water chemistry – corrosion interaction
INCOM		
Pertti Aaltonen, MSc	VTT TUO	INCOM sub-project manager
Kari Mäkelä, DTech	VTT TUO	Water chemistry – corrosion interaction
Kari Lahdenperä, LicTech	VTT TUO	NDE of steamgenerator tubing
INPERF		
Kim Wallin, DTech	VTT TUO	Physics modelling of irradiation damages
Matti Valo, MSc	VTT TUO	Damage mechanisms
Pertti Aaltonen, MSc	VTT TUO	Integrity of reactor internals
INCOORD		
Pentti Kauppinen, DTech	VTT TUO	INTELI project manager, coordination of INTELI-research

LWR oxide model for improved understanding of activity build-up and corrosion phenomena (LWROXI) (2004)
Aktiivisuuden kerääntymisen ja korroosion mallintaminen LWR-olosuhteissa (2004)

Research organisation: VTT Industrial Systems
Project manager: Martin Bojinov, VTT Industrial Systems
Deputy project manager: Petri Kinnunen, VTT Industrial Systems

Person	Org.	Task
Martin Bojinov, PhD	VTT	Project manager, modelling of surface films, electrochemical experiments
Petri Kinnunen, DTech	VTT	Deputy project manager, electrochemical experiments, modelling of surface films

Ageing of the Function of the Containment Building (AGCONT)
Suojarakennustoiminnon ikääntyminen

Research organisation: VTT Building and Transport
Project manager: Kalervo Orantie, VTT Building and Transport
Deputy project manager: Erkki Vesikari, VTT Building and Transport

Person	Org.	Task
Kalervo Orantie, MScTech	VTT	Project manager, Ageing of the function on containment
Erkki Vesikari, MScTech	VTT	Deputy project manager

Participation in the OECD NEA Task Group Concrete Ageing (CONAGE) (2003)
Osallistuminen OECD NEA betonirakenteiden vanhentumistyöryhmän työskentelyyn (2003)

Research organisation: VTT Building and Transport
Project manager: Erkki Vesikari, VTT Building and Transport

Person	Org.	Task
Erkki Vesikari, MScTech	VTT	Project manager, Participation in the OECD NEA Task Group Concrete Ageing (CONAGE)

Concrete Technological Studies Related to the Construction, Inspection and Reparation of the Nuclear Power Plant Structures (CONTECH)
Ydinvoimalarakenteiden rakentamiseen, tarkastamiseen ja korjaamiseen liittyvät betoniteknilliset tutkimukset

Research organisation: VTT Building and Transport
Project manager: Liisa Salparanta, VTT Building and Transport
Deputy project manager: Erkki Vesikari, VTT Building and Transport

Person	Org.	Task
Liisa Salparanta, MScTech	VTT	Project manager
Erkki Vesikari, LicTech	VTT	Research Scientist
Heikki Kukko, Dr Tech	VTT	Research Scientist
Pertti Pitkänen, MSc Tech	VTT	Research Scientist
Hannele Kuosa, MSc Tech	VTT	Research Scientist
Kyösti Laukkanen, MSc Tech	VTT	Research Scientist

Wall response to soft impact (WARSI)
Lentokonetörmäykset

Research organisations: VTT Industrial Systems (TUO), VTT Processes (PRO), Technical University of Tampere (TUT) and Radiation and Nuclear Safety Authority (STUK)
Project manager: Arja Saarenheimo, VTT Industrial Systems
Deputy project manager: Kim Calonius, VTT Industrial Systems

Persons	Organisation	Task
Arja Saarenheimo, Senior Research Scientist	VTT Industrial Systems	Project managing, precalculations for impact tests
Kim Calonius, Research Scientist	VTT Industrial Systems	ABAQUS analyses
Ari Vepsä, Trainee Research Scientist	VTT Industrial Systems	
Hannu Martikainen, Research Scientist	VTT Industrial Systems	LSDYNA analyses
Ari Aalto, Senior Assistant	Tampere University of Technology	Deformable missile, analyses related to loading transients due to impacts
Markku Tuomala, Professor	Tampere University of Technology	
Erik Eriksson, Student	Tampere University of Technology	M.Sc thesis on deformable missile
Ari Silde	VTT Processes	Droplet formation of liquid fuel

Impact tests (IMPACT)
Lentokonetörmäyksen kokeellinen simulointi

Research organisations:VTT Building and Transport (RTE)
Project manager: Tuomo Kärnä, VTT Building and Transport

Person	Org.	Task
Tuomo Kärnä, DTech	VTT	Project manager, Experimental apparatus, Models of the impacting objects and the reaction wall, Impact tests, Data analysis
Heikki Haapaniemi, MScTech	Fortum Service	Impact tests, Data analysis
Ilkka Hakola, MScTech	VTT	Experimental apparatus, Impact tests, Data analysis
Jouni Hietalahti, research engineer	VTT	Experimental apparatus, Impact tests, Reaction wall construction
Juha Juntunen, MScTech	VTT	Experimental apparatus, Impact tests, Data analysis, High-speed photography
Jaakko Johansson, technician	VTT	Impact tests
Juha Kurkela, MScTech	VTT	Impact tests, Data analysis
Erkki Järvinen, MScTech	VTT	Impact tests, Data analysis
Pekka Laine, DI	VTT	Experimental apparatus
Leo Lapinluoma, technician	VTT	Experimental apparatus, Impact tests, Reaction wall construction
Ari Lehtonen, technician	VTT	Reaction wall construction
Ilkka Linna, research engineer	VTT	Experimental apparatus, Impact tests
Heikki Lintunen, research engineer	VTT	Reaction wall construction
Jukka Mäkinen, technician	VTT	Experimental apparatus , Impact tests, Data analysis
Lasse Mörönen, LicTech	VTT	Models of the impacting objects and the reaction wall
Pekka Nurkkala, MScTech	Fortum Service	Impact tests, Data analysis
Kyösti Ovaska, technician	VTT	Impact tests
Matti Pajari, DTech	VTT	Models of the impacting objects and the reaction wall

The Integration of Thermal-Hydraulics (CFD) and Structural Analyses (FEA) Computer Codes in Liquid and Solid Mechanics (MULTIPHYSICS)
Termohydrauliikka- ja rakenneanalyysikoodien linkittäminen neste-rakennesysteemissä

Research organisation: FNS, VTT, ENPRIMA
Project manager: Pentti Varpasuo, Fortum Nuclear Services Ltd
Deputy project manager: Timo Toppila, Fortum Nuclear Services Ltd

Person	Org.	Task
Pentti Varpasuo, DTech	FNS	Project manager
Timo Toppila, MScTech	FNS	Deputy project manager
Ville Lestinen, MScTech	FNS	
Timo Pättikangas, DTech	VTT Processes	
Antti Timperi, MScTech	VTT Industrial Systems	
Arja Saarenheimo, MScTech	VTT Industrial Systems	
Pekka Iivonen, MScCivEng	ENPRIMA	
Seppo Orivuori, LicTech	ENPRIMA	

The Integral Code for Design Basis Accident Analyses (TIFANY) 2003
Integroitu malli suunnittelun perustana olevien onnettomuuksien laskentaan

Research organisations: Fortum Nuclear Services Ltd and VTT Processes (PRO)
Project manager: Kai Salminen, Fortum Nuclear Services Ltd

Person	Org.	Task
Kai Salminen, MScTech	FNS	Project manager, APROS development
Petra Lundström, MScTech	FNS	Deputy project manager
Davit Danielyan, MScTech	FNS	APROS development
Erkki Eskola, MScTech	FNS	APROS development
Harri Kontio, MScTech	FNS	APROS development
Eerikki Raiko, MScTech	FNS	APROS development
Ari Silde, MScTech	VTT	APROS development
Markku Hänninen, MScTech	VTT	APROS development
Jukka Ylijoki, MScTech	VTT	APROS development

APROS modelling of containment pressure suppression systems (TIFANY) 2004
Suojarakennuksen paineenalennusjärjestelmien APROS-mallien kehittäminen

Research organisations: Fortum Nuclear Services Ltd and VTT Processes (PRO)
Project manager: Kai Salminen, Fortum Nuclear Services Ltd
Deputy project manager: Petra Lundström, Fortum Nuclear Services Ltd

Person	**Org.**	**Task**
Kai Salminen, MScTech	FNS	Project manager, APROS development
Petra Lundström, MScTech	FNS	Deputy project manager
Davit Danielyan, MScTech	FNS	APROS development
Tommi Henttonen, trainee	FNS	APROS development
Tomi Routamo, MScTech	FNS	APROS development
Ari Silde, MScTech	VTT	APROS development
Markku Hänninen, MScTech	VTT	APROS development
Juha Poikolainen, MScTech	VTT	APROS development
Jukka Ylijoki, MScTech	VTT	APROS development

Thermal hydraulic analysis of nuclear reactors (THEA)
Termohydrauliikka-analyysit

Research organisation: VTT Processes
Project manager: *Minna Tuomainen, VTT Processes*,* Ismo Karppinen, VTT Processes**
Deputy project manager: Ismo Karppinen, VTT Processes**

Person	**Org.**	**Task**
Heikki Holmström, MScTech	VTT	Follow-up of OECD PSB-VVER and USNRC/CAMP
Markku Hänninen, LicTech	VTT	Apros improvments
Jorma Jokiniemi, DrTech	VTT	Participation in OECD/GAMA Writing group for SOAR on Nuclear Aerosols
Ismo Karppinen, MScTech	VTT	*Deputy project manager* / Project manager**
Jarto Niemi, MScTech	VTT	Apros improvments, Development of methods for multidimensional two-phase flow simulations
Jaakko Miettinen, MScTech	VTT	Development of porous media code PORFLO
Juha Poikolainen, MScTech	VTT	Apros improvments
Risto Sairanen, DrTech	VTT	Follow-up of OECD GAMA (in 2003)
Minna Tuomainen, MScTech	*VTT*	*Project manager, follow-up of OECD GAMA (in 2004) and OECD PKL*

** resigned from VTT in December 2004*
*** project manager from December 2004*

Severe accidents and nuclear containment integrity (SANCY)
Vakavat reaktoronnettomuudet ja suojarakennuksen kestävyys

Research organisations: VTT Processes and VTT Industrial Systems (TUO)
Project manager: Ilona Lindholm, VTT Processes
*Deputy project manager: Jari Tuunanen, VTT Processes**

Person	**Org.**	**Task**
Ilona Lindholm, MScTech	VTT/PRO	Project manager, Uncertainties of severe accident uncertainties, OECD/MCCI follow-up
Stefan Holmström, LicTech	VTT/TUO	STYX experiments
Pekka H. Pankakoski, MScTech	VTT/TUO	STYX experiments
Ensio Hosio, technician	VTT/TUO	STYX experiments
Ismo Kokkonen, MscTech	Fortum NS	Permeability measurements of FeCr
Riitta Zilliacus, MSc	VTT/PRO	Seal material irradiation, GEMINI2 thermochemical analyses
Harri Joki, MScTech	VTT/PRO	Material testing/irradiated seal material
Tommi Kekki, MSc	VTT/PRO	GEMINI2 thermochemical analyses
Kari Ikonen, DTech	VTT/PRO	Mechnaical analysis of pressure vessel lower head of OL-1/2
Pekka Kanerva, Research Trainee	VTT/PRO	Development of water ingression model WATING
Jaakko Miettinen, LicTech	VTT/PRO	Development of water ingression model WATING
Tuomo Sevón, Research Trainee	VTT/PRO	Steam explosion literature study Core-concrete interaction modelling/CCI-1 test analysis

** Resigned from VTT in 2004*

Management of Fission Product Gases and Aerosols
Fissiotuotekaasujen ja aerosolien hallinta (FIKSU)

Research organisation: VTT Processes
Project manager: Jorma Jokiniemi, VTT Processes
Deputy project manager: Ari Auvinen, VTT Processes

Person	**Org.**	**Task**
Jorma Jokiniemi, PhD	VTT	Project manager
Ari Auvinen, MScTech	VTT	Deputy project manager, Participation in Phebus project
Ulrika Backman, MSc	VTT	Ruthenium experiments
Unto Tapper, PhD	VTT	Electron microscopy
Riitta Zilliacus, MSc	VTT	Chemical analysis
Maija Lipponen, MSc	VTT	Chemical analysis
Tommi Renvall, MScTech	VTT	Radio tracer measurements

Development of Aerosol Models to Nuclear Applications
Aerosolimallien kehittäminen ydinvoimasovelluksiin (AMY)

Research organisation: Fortum Nuclear Services Ltd., VTT Processes
Project manager: Tomi Routamo, Fortum Nuclear Services Ltd.

Person	Org.	Task
Tomi Routamo, MScTech	FNS	Project manager, source term model implementation, SaTu system development
Satu Siltanen, MScTech	FNS	SaTu system development
Ville Karttunen, student	FNS	APROS SA fission product model development
Jorma Jokiniemi, PhD	VTT	Resuspension model development
Ari Auvinen, MScTech	VTT	Resuspension experiments and model development

Emergency preparedness supporting studies (OTUS)
Onnettomuusvalmiuden tukiselvitykset

Research organisation: VTT Processes
Project manager: Jukka Rossi, VTT Processes

Person	Org.	Task
Jukka Rossi, MScTech	VTT	Project manager, Radiation levels in severe accidents during refuelling and maintenance outage, Literature review of sea breeze

Archiving experiment data (KOETAR)
Koetulosten arkistointi

Research organisation: Lappeenranta University of Technology
Project manager: Vesa Riikonen, Lappeenranta University of Technology
Deputy project manager: Markku Puustinen, Lappeenranta University of Technology

Person	Org.	Task
Vesa Riikonen, MScTech	LUT	Project manager, checking and archiving data
Markku Puustinen, MScTech	LUT	Deputy project manager, checking and archiving data

Condensation pool experiments (POOLEX)
Lauhdutusallaskokeet

Research organisation: Lappeenranta University of Technology
Project manager: Markku Puustinen, Lappeenranta University of Technology
Deputy project manager: Heikki Purhonen, Lappeenranta University of Technology

Person	Org.	Task
Markku Puustinen, MScTech	LUT	Project manager, experiment planning and analysis
Heikki Purhonen, LicTech	LUT	Deputy project manager, OECD planning, international tasks, experiments
Jani Laine, MScTech	LUT	Experiment analysis, data conversion
Vesa Riikonen, MScTech	LUT	Data acquisition, experiments
Antti Räsänen, Student	LUT	Instrumentation, data acquisition, visualization, control systems, experiments
Harri Partanen, Engineer	LUT	Designing of test facility, experiments
Hannu Pylkkö, Technician	LUT	Construction, operation and maintenance of test facility, experiments
Ilkka Saure, Technician	LUT	Construction, operation and maintenance of test facility, experiments
Juha Kurki, Technician	LUT	Construction, operation and maintenance of test facility, experiments

PACTEL OECD project planning (PACO) (2004)
PACTEL OECD projektin suunnittelu (2004)

Research organisation: Lappeenranta University of Technology
Project manager: Heikki Purhonen, Lappeenranta University of Technology
Deputy project manager: Markku Puustinen, Lappeenranta University of Technology

Person	Org.	Task
Heikki Purhonen, LicTech	LUT	Project manager
Markku Puustinen, MScTech	LUT	Deputy project manager
Virpi Kouhia, MScTech	LUT	Research engineer, APROS analyses

Interaction approach to development of control rooms (IDEC)
Valvomoiden käyttäjäkeskeinen kehittäminen

Research organisation: VTT Industrial Systems
Project manager: Olli Ventä, VTT Industrial Systems

Person	Org.	Task
Olli Ventä , DTech	VTT	Project manager
Leena Norros, PhD	VTT	Performance evaluation for system usability
Paula Savioja MScTech	VTT	Interface evaluation for system usability
Maaria Nuutinen MSc (Psych)	VTT	Method validation

Application Possibilities of Systematic Requirements Management in the Improvement of Nuclear Safety in Finland (APSReM) (2003)
Systemaattisen vaatimustenhallinnan soveltamismahdollisuudet ydinturvallisuuden parantamisessa Suomessa (2003)

Research organisation: RAMSE Consulting Oy
Project manager: Veli Taskinen

Person	Org.	Task
Veli Taskinen	RAMSE	Project manager
Markus Renlund	RAMSE	
Pekka Kähkönen	RAMSE	

Influence of RoHS -directive to reliability of electronics, Prestudy (ROVEL) (2004)
RoHS -direktiivin vaikutus elektroniikan luotettavuuteen, esitutkimus (ROVEL) (2004)

Research organisations: VTT Industrial Systems, Electronics Product Technology
Project manager: Hannu Hossi, VTT Industrial Systems
Deputy project manager: Antti Turtola, VTT Industrial Systems

Person	Org.	Task
Hannu Hossi, MScTech	VTT Industrial Systems	Project manager, Definition of the contents
Antti Turtola, MScTech	VTT Industrial Systems	Deputy project manager, Modifications due to EU RoHS – directive.
Helge Palmén, LicTech	VTT Industrial Systems	Participation in IEC TC45 standardisation work

Organisational culture and management of change (CulMa)
Organisaatiokulttuuri ja muutoksen hallinta ydinvoimalaitoksissa

Research organisation: VTT Industrial Systems
Project manager: Teemu Reiman, VTT Industrial Systems
Deputy project manager: Pia Oedewald, VTT Industrial Systems

Person	Org.	Task
Teemu Reiman, M.A. Psych.	VTT	Project manager; cultural assessments, development of CAOC methodology, (dissertation work)
Pia Oedewald, M.A. Psych.	VTT	Deputy project manager; case studies concerning management of change and competence of maintenance personnel
Jari Kettunen, M.A.	VTT	Researcher; cases concerning organizing of work at NPPs, outsourcing
Kari Laakso, DTech.	VTT	Maintenance expert
Reetta Kurtti, Cognitive science student	VTT	Research assistant, CULTURE-survey data analysis

Disseminating Tacit Knowledge in Organizations (DIAMOND) (2004)
TIMANTTI – Tietopääoman hyödyntäminen organisaatiossa (2004)

Research Organization: Helsinki University of Technology / BIT Research Centre
Project Manager: Niina Rintala, BIT Research Centre
Deputy project manager: Laura Hyttinen, Helsinki University of Technology

Person	Org.	Task
Niina Rintala, LicTech	HUT	Project manager
Laura Hyttinen, LicTech	HUT	Researcher
Eila Järvenpää, DTech, Prof	HUT	Project leader

Potential of Fire Spread (POTFIS)
Palon leviämisen mahdollisuus

Research organisation: VTT Building and Transport
Project manager: Olavi Keski-Rahkonen, VTT Building and Transport
Deputy project manager: Johan Mangs, VTT Building and Transport

Person	Org.	Task
Olavi Keski-Rahkonen, DTech	VTT	Project manager, POTFIS
Johan Mangs, PhD	VTT	Deputy project manager, POTFIS

Principles and Practices of Risk-Informed Safety MAnagement (PPRISMA)
Riskitietoisen turvallisuudenhallinnan periaatteet ja käytännöt

Research organisation: VTT Industrial Systems
Project manager: Jan-Erik Holmberg, VTT Industrial Systems
Deputy project manager: Urho Pulkkinen, VTT Industrial Systems

Person	Tasks
Pentti Haapanen, MScTech	Reliability of computer-based systems
Atte Helminen, MScTech	Reliability of computer-based systems
Jan-Erik Holmberg, DTech	Project manager, risk-informed ways of management of fire situations, risk-informed classification, human reliability, international co-operation
Kristiina Hukki, MA	Risk-informed ways of management of fire situations
Mika Koskela, MScTech	Reliability of computer-based systems
Kari Laakso, DTech	Maintenance and operability strategies
Eija Myötyri, MScTech	Risk-informed classification, reliability of computer-based systems
Ilkka Männistö, MSc student	Risk-informed classification
Urho Pulkkinen, DTech	Assistant project manager (2004), risk-informed decision making, risk-informed classification, reliability of computer-based systems
Tony Rosqvist, DTech	Maintenance and operability strategies
Kaisa Simola, DTech	Assistant project manager (2003), risk-informed decision making, international co-operation

Administration and information of the research programme (SAHA)
Tutkimusohjelman hallinto ja tiedotus

Research organisation: VTT Processes
Project manager: Eija Karita Puska, VTT Processes
Deputy project manager: Hanna Räty, VTT Processes

Person	Org.	Task
Eija Karita Puska, DTech	VTT	Programme leader
Hanna Räty, MScTech	VTT	Project co-ordinator
Heikki Holmström, MScTech	VTT	EU FP6 national follow-up (HILJA database)
Timo Vanttola, DTech	VTT	JSRI-II EU-project (data base) (2003)

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

Abstract

SAFIR 2003–2006 is the present Finnish public research programme on nuclear power plant safety. The programme is administrated by the steering group that has been nominated by the Ministry of Trade and Industry (KTM). The steering group of SAFIR consists of representatives from Radiation and Nuclear Safety Authority (STUK), Ministry of Trade and Industry (KTM), Technical Research Centre of Finland (VTT), Teollisuuden Voima Oy (TVO), Fortum Oyj (Fortum), National Technology Agency of Finland (Tekes), Helsinki University of Technology and Lappeenranta University of Technology.

The six key research areas of SAFIR are 1) reactor fuel and core, 2) reactor circuit and structural safety, 3) containment and process safety functions, 4) automation, control room and IT, 5) organisations and safety management and 6) risk-informed safety management.

The research programme of the year 2004 involved 23 research projects, whose volume varied from a few person months to several person years. The total volume of the programme in 2004 was 4.9 million euros and 35 person years.

The research in the programme is performed primarily by the Technical Research Centre of Finland (VTT). Other research units responsible for the projects include Lappeenranta University of Technology, Fortum Nuclear Services Ltd and Helsinki University of Technology. In addition, there are a few minor subcontractors in some projects.

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

This report gives a summary of the results of the SAFIR programme for the period January 2003 – October 2004. During this period the programme produced 256 publications, four Doctoral, one Licentiate and six Master Thesis. The total volume of the programme in 2003–2004 was approximately 9 M€ and 67 person years.

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

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The six key research areas of SAFIR are 1) reactor fuel and core, 2) reactor circuit and structural safety, 3) containment and process safety functions, 4) automation, control room and IT, 5) organisations and safety management and 6) risk-informed safety management.

The research in the programme has been performed primarily by VTT. Other research units responsible for the projects include Lappeenranta University of Technology, Fortum Nuclear Services Ltd, Helsinki University of Technology and RAMSE Consulting. In addition, there have been a few minor subcontractors in some projects.

This report gives a summary of the results of the SAFIR programme for the period January 2003 – October 2004.

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