VTT TIEDOTTEITA – MEDDELANDEN – RESEARCH NOTES 1856

RETU
The Finnish Research Programme on Reactor Safety
Interim Report 1995 - May 1997

Edited by

Timo Vanttola & Eija-Karita Puska

VTT Energy

TECHNICAL RESEARCH CENTRE OF FINLAND
ESPOO 1997

ISBN 951-38-5146-X
ISSN 1235-0605

ISBN 951-38-5165-6 (URL: http://www.inf.vtt.fi/pdf/)
ISSN 1455-0865 (URL: http://www.inf.vtt.fi/pdf/)

JULKAISIJA – UTGIVARE – PUBLISHER

Valtion teknillinen tutkimuskeskus (VTT), Vuorimiehentie 5, PL 2000, 02044 VTT
puh. vaihde (09) 4561, faksi (09) 456 4374

Statens tekniska forskningscentral (VTT), Bergsmansvägen 5, PB 2000, 02044 VTT
tel. växel (09) 4561, fax (09) 456 4374

Technical Research Centre of Finland (VTT), Vuorimiehentie 5, P.O.Box 2000, FIN–02044 VTT, Finland
phone internat. + 358 9 4561, fax + 358 9 456 4374

VTT Energia, Ydinenergia, Tekniikantie 4 C, PL 1604, 02044 VTT
puh. vaihde (09) 4561, faksi (09) 456 5000

VTT Energi, Kärnkraft, Teknikvägen 4 C, PB 1604, 02044 VTT
tel. växel (09) 4561, fax (09) 456 5000

VTT Energy, Nuclear Energy, Tekniikantie 4 C, P.O.Box 1604, FIN-02044 VTT Finland
phone internat. + 358 9 4561, fax + 358 9 456 5000

Technical editing Leena Ukskoski

LIBELLA PAINOPALVELU OY, ESPOO 1997

Vanttola, Timo & Puska, Eija-Karita (eds.). RETU. The Finnish Research Programme on Reactor Safety. Interim Report 1995 - May 1997 (Reaktoriturvallisuuden tutkimusohjelma RETU. Puoliväliraportti 1995 - toukokuu 1997). Espoo 1997, Technical Research Centre of Finland, VTT Tiedotteita - Meddelanden - Research Notes 1856. 168 p. + app. 51 p.

UDC 621.039.58
Keywords nuclear power plants, nuclear reactors, reactor safety, nuclear reactor cores, accident prevention

Abstract

The Finnish national research programme on *Reactor Safety (RETU, 1995-1998)* concentrates on the search of safe limits of nuclear fuel and the reactor core, accident management methods and risk management of the operation of nuclear power plants. The annual volume of the programme has been about 26 person years and the annual funding FIM 15 million. This report summarises the structure and objectives of the programme, research fields included and the main results obtained during the period 1995 – May 1997.

In the field of *operational margins* of a nuclear reactor, the behaviour of high burnup nuclear fuel is studied both in normal operation and during power transients. The static and dynamic reactor analysis codes are developed and validated to cope with new fuel designs and complicated three-dimensional reactivity transients and accidents. The range of the models is extended by implementing advanced flow models and numerical solution methods.

Research on *accident management* aims at development and validation of calculation methods needed to plan preventive measures and to train the personnel to severe accident mitigation. Other goals are to reduce uncertainties in phenomena important in severe accidents and to study actions planned for accident management. The programme includes own experimental work, but participation in large international test programmes is of major importance. The Finnish thermal-hydraulic test facility PACTEL (Parallel Channel Test Loop) is used extensively for the evaluation of the VVER-440 plant accident behaviour, for the validation of the accident analysis computer codes and for the testing of passive safety system concepts for future plant designs.

In the field of *risk management* probabilistic methods are developed for safety related decision making and for complex phenomena and event sequences. Effects of maintenance on nuclear power plant safety are studied and more effective methods for the assessment of human reliability and safety critical organisations are searched. In order to enhance human competencies in the control of complex environments, practical tools for training and continuous learning are worked out, and methods to evaluate appropriateness of control room design are developed.

Vanttola, Timo & Puska, Eija-Karita (toim.). RETU. The Finnish Research Programme on Reactor Safety. Interim Report 1995 - May 1997 (Reaktoriturvallisuuden tutkimusohjelma RETU. Puoliväliraportti 1995 - toukokuu 1997). Espoo 1997, Valtion teknillinen tutkimuskeskus, VTT Tiedotteita - Meddelanden - Research Notes 1856. 168 s. + liitt. 51 s.

UDK 621.039.58
Avainsanat nuclear power plants, nuclear reactors, reactor safety, nuclear reactor cores, accident prevention

Tiivistelmä

Kansallisen *Reaktoriturvallisuuden tutkimusohjelman (RETU, 1995 - 1998)* keskeisiä alueita ovat polttoaineen ja reaktorisydämen turvallisten toimintarajojen määrittäminen, onnettomuustilanteiden hallinta sekä ydinvoimalaitoksen käytön riskien hallinta. Pääosin Valtion teknillisessä tutkimuskeskuksessa toteutettavan ohjelman vuotuinen laajuus on noin 26 henkilötyövuotta ja rahoitus 15 milj. mk. Raportissa on yhteenveto tutkimusalueista ja saavutuksista ohjelman alkujaksolta 1995 - toukokuu 1997.

Ydinpolttoaineen ja reaktorisydämen turvalliset toimintarajat -alueella suuripalamaisen ydinpolttoaineen käyttäytymistä tutkitaan sekä normaaliolosuhteissa että mahdollisten tehohäiriöiden aikana. Staattisia ja dynaamisia reaktorianalyysiohjelmistoja kehitetään ja kelpoistetaan kattamaan uudet polttoainetyypit sekä monimutkaiset kolmidimensioiset reaktiivisuushäiriöt ja onnettomuustilanteet. Mallien toiminta-alaa laajennetaan ottamalla käyttöön uusia virtauskuvauksia ja numeerisia ratkaisumenetelmiä.

Onnettomuustilanteiden hallinnalla pyritään vaaratilanteiden varhaiseen havaitsemiseen ja pysäyttämiseen sekä seurausten rajoittamiseen niin vähäisiksi kuin mahdollista. Alan tutkimuksessa kehitetään ja kelpoistetaan laskentamenetelmiä, joita tarvitaan ennaltaehkäisevien toimien suunnitteluun ja henkilöstön koulutukseen vakavien onnettomuuksien seurausten lieventämiseksi. Vakavien onnettomuuksien kannalta tärkeiden ilmiöiden epävarmuuksia pyritään pienentämään ja suunniteltuja onnettomuuden hallintamenetelmiä arvioidaan. Tutkimukseen sisältyy omaa kokeellista työtä, minkä lisäksi osallistutaan kansainvälisiin yhteishankkeisiin. Suomalaista PACTEL-laitteistoa (Parallel Channel Test Loop) käytetään laajasti VVER-440-laitoksen onnettomuuskäyttäytymisen arviointiin, onnettomuustilanteiden laskentaohjelmistojen kelpoistukseen ja uusien laitostyyppien passiivisten turvallisuuspiirteiden testaukseen.

Riskien hallinnassa kehitetään todennäköisyyspohjaisia menetelmiä turvallisuuteen liittyvälle päätöksenteolle sekä monimutkaisille ilmiöille ja tapahtumille. Huollon vaikutuksia ydinvoimalaitoksen turvallisuuteen tutkitaan sekä haetaan tehokkaampia tapoja arvioida inhimillistä luotettavuutta ja turvallisuuskriittisiä organisaatioita. Monimutkaisen ympäristön hallintaan kehitetään käytännöllisiä koulutusmenetelmiä ja edistetään jatkuvaa oppimista. Lisäksi luodaan valvomoiden toimivuuden arviointimenetelmiä.

Foreword

Based on the Finnish expert group report *Nuclear Energy Research until 2000 (1994),* the Ministry of Trade and Industry (KTM) launched two new national research programmes on the safety of nuclear reactors for the period 1995 - 1998. The new programmes, *Reactor Safety (RETU*) and *Structural Integrity of Nuclear Power Plants (RATU2)*, continue research in most areas of the completed programmes on *Nuclear Power Plant Systems Behaviour and Operational Aspects of Safety (YKÄ)* and *Nuclear Power Plant Structural Safety (RATU)*. This interim progress report gives an overview of the scientific and technical results obtained during the first two years of the RETU programme.

The Ministry of Trade and Industry has contracted the Technical Research Centre of Finland (VTT) to co-ordinate the programme. VTT also performs most of the research. The funding sources are KTM, VTT, the Finnish Centre for Radiation and Nuclear Safety (STUK), the Lappeenranta University of Technology, the nuclear power companies Imatran Voima Oy (IVO) and Teollisuuden Voima Oy (TVO) and the European Union.

In addition to the editors, several people have contributed to the preparation of this report, in particular the project managers Mr. Kari Ranta-Puska[1], Mr. Risto Teräsvirta[2], Ms. Hanna Räty[1], Dr. Jari Tuunanen[1], Mr. Risto Sairanen[1], Mr. Pekka Pyy[3] and Dr. Leena Norros[3]. Ms. Anne Marttila[1] carefully finalized the layout of the report.

The editors are grateful to all the colleagues participating the work of the programme and/or assisting in preparing this report, as well as to the members of the steering and reference groups of the programme for providing their experience in the planning and steering of the research.

---

[1] VTT Energy
[2] IVO Power Engineering Ltd
[3] VTT Automation

Contents

1. Introduction

Goals for nuclear energy research in Finland

The focus of nuclear research work in Finland is on the safety and operational performance of the power plants, and on the management and disposal of radioactive wastes. The current research policy for publicly funded nuclear energy research rests on the following premises:

- The operation of the existing nuclear plants will be continued as long it is safe and economically sound. Continued R&D aims at improvements in safety, operational performance and economics. Plant lifetime extension objectives are considered in planning the maintenance and refurbishment work.

- Research in nuclear waste management is geared to the schedule set in the government's programme plan of 1983, as refined by later decisions.

- Expertise is maintained also for the possible expansion of nuclear power capacity.

In order to make publicly funded nuclear energy research more result-oriented and efficient, most of it is structured in nationally coordinated research programmes. The programmes are organized in such a way that they also support the basic and advanced education of experts and facilitate systematic, efficient international cooperation. The three research areas and the programmes recently completed or under way are:

<u>Operational safety</u>: Systems behaviour and operational aspects of safety (YKÄ, 1990-1994) and Reactor safety (RETU, 1995-1998), described in this report

<u>Structural safety</u>: Nuclear power plant structural safety (RATU, 1990-1994) and RATU2 (1995-1998)

<u>Waste management</u>: Publicly financed research on nuclear waste management (JYT, 1989-1993), JYT2 (1994-1996) and JYT2001 (1997-2001).

The Ministry of Trade and Industry (KTM) uses about 75 % of its nuclear energy research budget directly on these three programmes. Other major contributors are the Technical Research Centre of Finland (VTT) and the Finnish Centre for Radiation and Nuclear Safety (STUK). The power companies, Imatran Voima Oy (IVO) and Teollisuuden Voima Oy (TVO) also participate in the funding of many of the projects. In addition to the nuclear fission energy research, there is a parallel national research programme on nuclear fusion

(FFUSION) in progress, funded by the Technology Development Centre Finland (TEKES) and coordinated by VTT.

Publicly funded nuclear fission energy research is particularly intended to support decision-making in energy policy and to provide independent and impartial expertise for the regulation of nuclear energy. The public sector has a major role in providing the necessary education, personnel, and equipment resources for research and development, and in establishing the framework for international information exchange and co-operation.

The nuclear regulatory authorities need expertise based on research when supervising the use of nuclear energy. Their main tasks are to develop safety criteria for the design of nuclear power plants, to supervise the construction, refurbishment and operation of the power plants and to establish efficient emergency plans for abnormal situations. Furthermore, the technical and financial plans of the utilities for the management and disposal of nuclear waste are to be submitted for the review and approval of the nuclear regulatory authorities.

Organisation of the research activities

In Finland nuclear energy research and development has been decentralized from the beginning. Large part of the research takes place in the various research institutes of VTT (Fig. 1). The other major research institutions are the University of Helsinki, the Universities of Technology in Helsinki and Lappeenranta (TKK, LTKK), the Geological Survey of Finland (GTK) and the Meteorological Institute. In addition, the reactor safety authority (STUK) and the power companies IVO and TVO carry out own internal research.

The Advisory Committee on Nuclear Energy (YEN) assists KTM in guiding and directing the publicly funded nuclear energy research. In addition, KTM has appointed a steering group for each research programme to supervise and direct the research.

Extensive worldwide international collaboration is characteristic of nuclear research. The importance of international research for Finland is most evident in the area of large-scale experimental research, collection of basic nuclear data and in the development of large computer programs, especially those used in safety analyses.

Overall view of nuclear energy research in Finland

Fig. 2 provides an overall view of the directions and funding sources of nuclear energy research in Finland in 1996. The total volume was about FIM 180 million, which corresponds to over 200 person years. The figures are somewhat higher than during normal years because of the ongoing intensive modernisation projects of the Finnish nuclear power plants. About two thirds of the activities were funded by the utilities. The national research programmes cover about 1/4 of the total research activities. The largest single research effort is the utility-funded waste management programme. In the reactor safety field, the national research programmes represent about 1/5 of the total. The rest mostly includes plant-specific research on reactor materials, severe accident management and thermal-hydraulics, carried out mostly by the utilities. During 1996 this sector was particularly large because of research efforts of the utilities in the modernisation projects. The internal research conducted by STUK mainly focuses on environmental impacts and radiation protection.

VTT Activities in Nuclear Energy R&D

VTT Chemical Technology
* Analytical Chemistry of Nuclear Materials
* Chemistry of Waste Disposal
* Aerosol Techniques
* Nuclear Measurement Techniques
* Research Reactor

VTT Automation
* Supervision and Control Systems
* PSA
* Human Factors
* Process Simulation

VTT Energy

* Reactor Safety
* Fuel Management
* Waste Management
* Fusion Technology

* Economics of Nuclear Power

* Aerosol Technology
* Process Simulation

VTT Building Technology
* Fire Technology

VTT Manufacturing Technology
* Materials Ageing
* Radiation Embrittlement
* Structural Integrity
* In-Service Inspections

VTT Communities and Infrastructure
* Rock and Backfill Performance of Waste Repositories
* Social Impact Assessment of Nuclear Waste Management

Figure 1. VTT activities in nuclear energy R&D.

Structure of this report

Chapter 2 provides an overall description of the research programme on *Reactor Safety (RETU 1995-1998).* Chapter 3 describes in detail the state of research and the achievements of the research projects. Chapter 4 summarises the intermediate results. The report is supplemented with annexes for international co-operation, publications, list of persons in the programme management and staff, academic degrees awarded and project research plans for the rest of the programme (1997 - 1998).

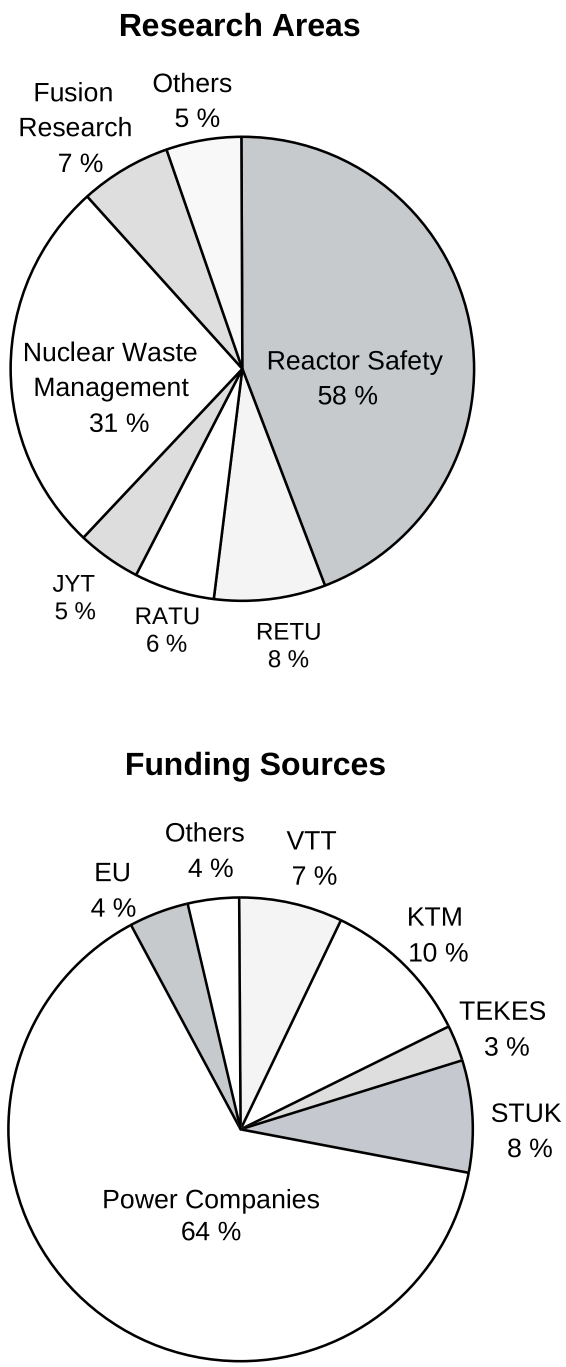


Figure 2. Direction and funding distribution of nuclear energy research in Finland in 1996. The total annual funding is about FIM 180 Million and the research volume is some 200 person years. Research area "Others" includes environmental impacts, research reactor and radiation protection.

2. Description of the research programme

Programme scope, objectives and structure

The Finnish national research programme on *Reactor Safety (RETU, 1995-1998)* concentrates on the search of safe limits of nuclear fuel and the reactor core, accident management methods and risk management of the operation of nuclear power plants.

The programme was planned in co-operation by the nuclear regulatory authority STUK, the nuclear utilities and the research institutions in a working group set up by KTM.

The organisation of the research programme is described in Fig. 3. The members in various bodies of the programme appear in Annex C.

The research field covered is summarised in Fig. 4. Table 1 lists the projects and provides some overall information. Based on the initial overall framework of the programme, detailed work plans for the projects are formulated annually (Annex E). The project structure is flexible in order to accommodate changing priorities and to consider the feedback obtained.

Duration:	1995-1998
Total funding:	FIM 55-65 mill.
KTM funding:	FIM 20-22 mill.
Volume in 1997:	FIM 15.9 mill. 27 person years
Funding in 1997:	
KTM	FIM 5.2 mill.
VTT	FIM 5.4 mill.
STUK	FIM 1.5 mill.
Utilities	FIM 1.3 mill.
EU/NFS-2	FIM 1.7 mill.
Others	FIM 0.8 mill.
Research institutions:	
VTT	
Lappeenranta Univ. Tech. (LTKK)	
Imatran Voima Oy (IVO)	
Programme management:	
VTT Energy	
Manager: Dr. Timo Vanttola	

General objectives of the programme

- Develop tools and practices for safety authorities and utilities
- Support the identification, assessment and implementation of safety improvements
- Provide a basis for the validation of safety-related decisions
- Provide favourable conditions for the education of nuclear experts
- Promote technology transfer

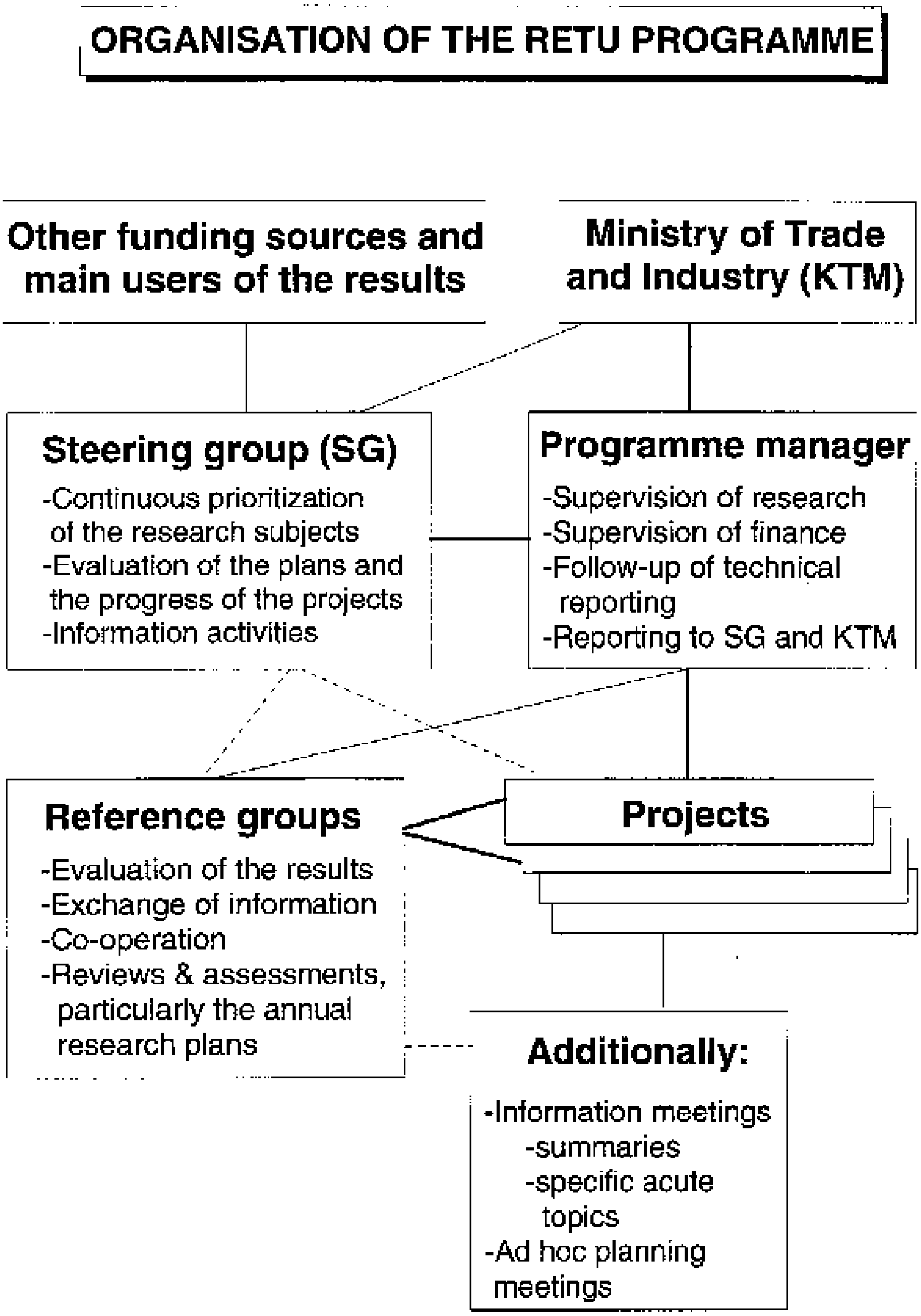


Figure 3. Organisation of the RETU research programme.

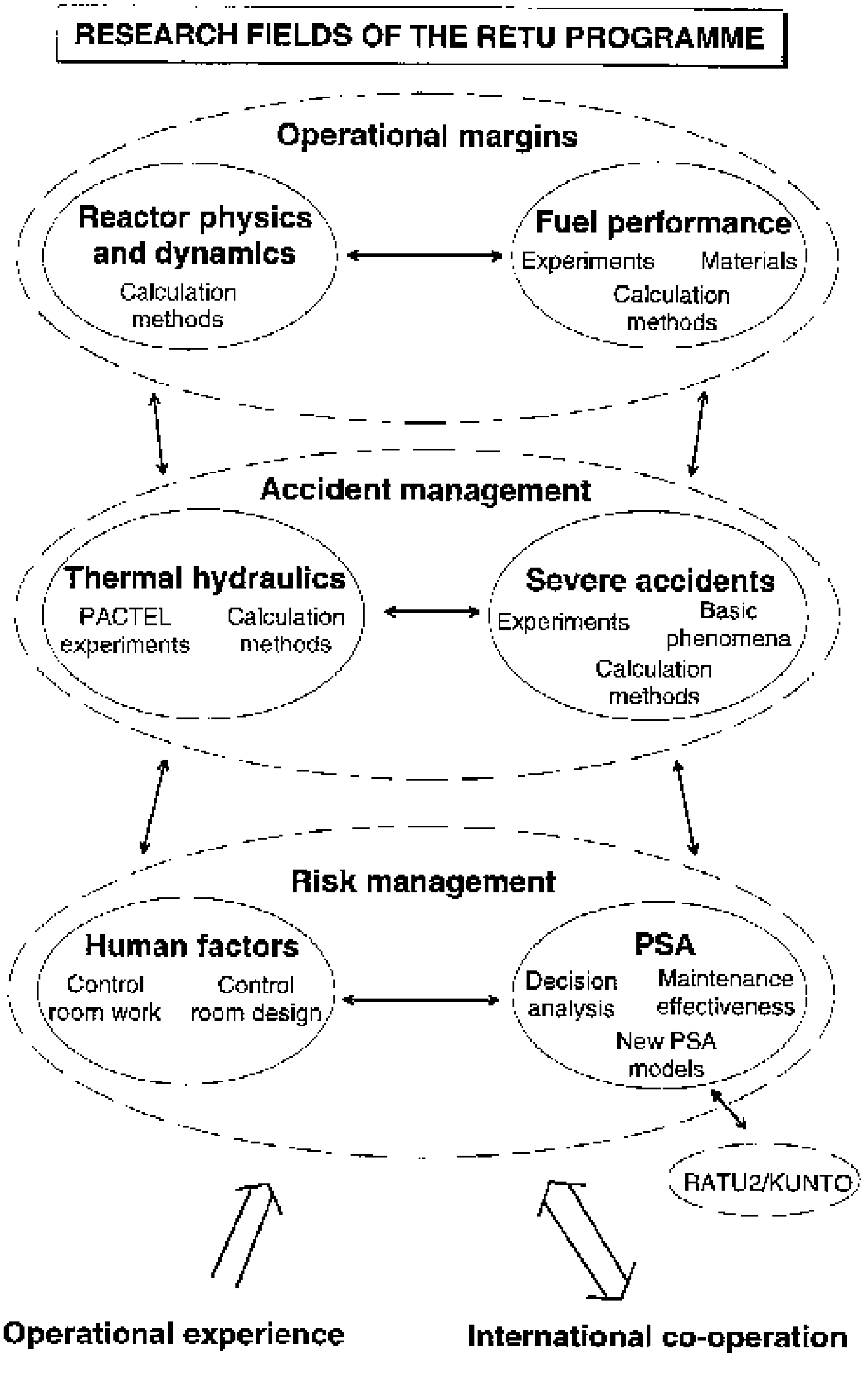


Figure 4. The research fields of the RETU research programme.

Table 1. Projects of the RETU research programme.

Research field Project (Acronym)	Project manager	Duration (years)	Participating institutions	Volume (person years) -95	 -96	 -97 pl.
Operational margins of nuclear fuel and reactor core:						
Transient models of nuclear fuel (PATRA)	Mr. Kari Ranta-Puska	94 –97	VTT Energy	1.7	1.3	1.5
VVER fuel experiments (SOFIT) (Reorganized and removed from public programme in 1996)	Mr. Risto Teräsvirta	85-95	IVO	1.0	-	-
Calculation methods of reactor physics and dynamics (DYNAMIC)	Ms. Hanna Räty	95 - 98	VTT Energy	4.1	5.9	5.2
Assessment of complex transient and accident situations / Accident management::						
Thermal hydraulic experiments and analyses (TEKOJA)	Dr. Jari Tuunanen	95 - 98	VTT Energy and LTKK	8.6	4.9	4.6
Passive safety injection experiments (PAHKO)	Dr. Jari Tuunanen	96-98	VTT Energy and LTKK	-	3.9	3.3
Severe accident management (VAHTI)	Mr. Risto Sairanen	94 - 96	VTT Energy and IVO	6.8	6.6	-
Reactor accidents' phenomena and simulation (ROIMA)	Mr. Risto Sairanen	97 - 99	VTT Energy and IVO	-	-	8.1
Management of nuclear power plant risks:						
Reliability and risk analyses (LURI)	Mr. Pekka Pyy	95-98	VTT Automation	2.0	2.3	2.4
Human factors in NPP operations (ORINT)	Dr. Leena Norros	95-98	VTT Automation	1.1	0.8	1.2
Administration of RETU:						
Administration and information activities (REHTI)	Dr. Timo Vanttola	95-98	VTT Energy	0.7	0.5	0.5
				26.0	26.2	26.8

The scope and objectives of the research in the various fields of the programme are briefly summarised below.

Operational margins of nuclear fuel and reactor core

Fuel behaviour research concentrates on validating fuel behaviour codes for higher burnups and on obtaining knowledge on improved and additional material properties of nuclear fuels used in the Finnish reactors. Particularly methods applicable to fast transients, such as reactivity initiated accidents are developed, acquired and validated. Accuracy and application range of the steady state model under normal use is also continuously evaluated.

Code development in the fields of reactor physics and dynamics, and in related thermal-hydraulics, has been one of the key areas of reactor safety research in Finland, particularly for the needs the Loviisa VVER plant. The development work includes implementation of the latest versions of reactor physics codes and data libraries, construction of 3-D core dynamics models, improved flow models and numerical solution techniques. Recently major tasks have been extension, validation and application of the dynamics codes to complex reactivity transients, such as ATWS, boron dilution events and BWR core stability.

Assessment of complex transient and accident situations / Accident management

Various aspects of reactor accidents are studied in order to support the accident management plans of the Finnish nuclear power plants. Thermal-hydraulic computer codes are developed and validated. Severe accident computer codes acquired through international co-operation are adapted and validated for the Olkiluoto and Loviisa plants. A structural mechanics code, developed to understand the behaviour of the reactor pressure vessel in severe accidents, is mainly devoted to failure mechanisms of the lower head penetrations of the pressure vessel (BWR). In order to understand fission product behaviour in the containment building, aerosol experiments are conducted in various test facilities using simulant materials. A training tool for severe accidents is developed by implementing selected severe accident models into a simulator environment.

In order to study VVER specific accident behaviour, thermal-hydraulic tests are performed with the PACTEL facility (PArallel Channel TEst Loop), that is scaled 1:1 vertically and 1:305 in volume to the Loviisa plant. The goals of these experiments are to understand the basic thermal-hydraulic phenomena in a VVER cooling system, to study effectiveness of various accident management measures and to generate data for code validation. The same facility is also used for testing passive safety injection of emergency core cooling proposed for advanced reactor concepts in a project funded by EU.

Management of nuclear power plant risks

Probabilistic methods are developed for nuclear safety related decisions and for modelling complex phenomena and event sequences. The effects of maintenance on nuclear power plant safety are studied and more effective methods are searched for the assessment of human reliability and safety critical organisations. As a part of the work, differences in safety culture in the Nordic countries have been explored.

Safe operation of power plants is also enhanced through development of human competencies in the control of complex environments. The competencies are studied from the decision making point of view. The methods developed for the analysis of decision making are further modified to provide practical tools for training and continuous learning, design and validation of control room design and control of safety.

3. Scientific and Technical Results

3.1 LWR fuel performance

Extending the fuel discharge burnup continues to be a global trend in search of improved economy of the LWR fuel cycle. For increased burnups, different management schemes and new fuel types are being introduced. This, on the other hand, sets new requirements on the fuel research because of the numerous phenomena that depend on the fuel rod design parameters and irradiation conditions. The reliability and safety of fuel utilisation must be preserved in all circumstances.

In the preceding fuel research project POTRA (Limits of safe fuel operation), fuel performance and materials data under steady-state operation were collected and qualified in parallel with the development and validation of fuel performance codes. A modern fuel performance model, ENIGMA, was acquired by VTT with financial support from the Finnish utilities. The code was benchmarked against test reactor results from international programmes and pool-side examinations of the Finnish reactors. The results of these assessments give high confidence on the code's capability in studying the limits of fuel performance, on best-estimate basis.

The current fuel research programme PATRA was launched at VTT to address questions of high burnup fuel performance in transient and accident conditions. Especially French and Japanese tests simulating conditions of a fast Reactivity Initiated Accident (RIA) at progressively higher burnups, have indicated a marked decrease in the peak enthalpy threshold for cladding failure. The failure mechanism may vary depending on the type of the RIA, and the material characteristics seem to have an important role, as well. The uncertainties involved in the fuel behaviour have raised serious concern, and it may become necessary to re-evaluate the acceptance criteria that are based on results from fresh or low burnup fuel experiments. The objectives of the PATRA project were to acquire the essential knowledge for assuring the safety of a high burnup fuel in transient conditions and to update the corresponding fuel performance models.

3.1.1 Fuel performance under RIA, acquiring the necessary knowledge through international co-operation

A complex of problems related to the high burnup fuel RIA behaviour has been identified word-wide, but the relative significance of the phenomena has not been resolved. Experimental work on this subject is mostly performed in France (Cabri test reactor), in Japan (NSRR) and in Russia (IGR). In the PATRA-project, first goal was to review the available experience and take part in international co-operation.

Review of the failure mechanisms

In reactor operation, the properties of cladding material will be changed due to neutron bombardment. With increasing neutron fluence the cladding becomes less ductile while its strength increases. Oxide thickness of the outer surface increases with increasing burnup essentially in a linear way up to certain burnup (~40 MWd/kgU in LWR's). Thereafter, accelerated corrosion is often observed in conventional Zircaloy claddings. Together with the thicker oxide layer, the hydrogen uptake of the cladding will increase up to 600 to 700 ppm with an oxide thickness of 100 μm. In the cladding, a high hydride concentration causes loss of ductility and a decrease in tensile strength. If loss of oxide or "spalling" takes place, a cooler and even more favourable region is formed for hydride build-up and PCMI failure. Thus hydride assisted brittle failure may be expected early into a fast RIA-transient, when the pellet has reached a high temperature, but the cladding is still close to the normal operation temperature.

If the cladding does not fail, while relatively cool, during the early stage of the energy insertion, a possible failure mechanism can be bursting through ballooning, rather than a brittle fracture. Above about 650 ^{o}C, Zircaloy becomes ductile again while radiation and hydride assisted embrittlement are annealed out. This may be a more possible failure mechanism for claddings with less oxide and hydride or in the association with wider pulses, like hundred milliseconds or more in half-peak width. Cladding materials with low tin content and niobium alloying (like VVER Zr1%Nb cladding) show better resistance to corrosion, and thus hydride assisted brittle fracture may be less probable even at high burnup. It is expected therefore, that the standard Zircaloy materials are more susceptible to failures at low enthalpies than the "low tin" or zirconium niobium alloys, which fail more probably by mechanisms that relate to high temperatures and to higher inserted energies.

In search of the failure limits the detailed role of various parameters has not been clarified yet. Neither the pulse height nor the total inserted enthalpy alone, but also the pulse width may be an important factor. In addition, the failure mode can strongly depend on the coolant conditions and response.

International co-operation

A Task Force of the Nuclear Science Committee (NSC) of OECD/NEA finalised a report on the key issues of the fuel behaviour that need further investigations in order to reach good predictive capabilities at extended burnup (OECD 1995). In the Finnish contribution attention was paid on the issue of reactivity initiated accident behaviour. To enlighten the subject a large international meeting was organised later. The role of burnup effects of fuel properties in RIA analyses was discussed in a paper presented by VTT Energy (Kyrki-Rajamäki 1995). For the first time, the degradation of UO_2 thermal conductivity with burnup was taken into account in the reactor dynamics calculation, and as a result, one concluded that the highest enthalpies were not always found in assemblies with the highest power densities.

An agreement on the co-operation in the field of nuclear safety was reached between the French IPSN (Institute de Protection et de Sûreté Nucléaire) and VTT (MOU 1995). Specific topics that concern the PATRA project include collaboration on the studies of highly irradiated fuel under reactivity initiated accidents and the French SCANAIR computer code development. The agreement gives VTT a right to use the SCANAIR code Version 2.2, and, in return, VTT Energy will apply multidimensional mechanical approach in the calculation of fuel rod deformations under specific RIA tests and participate in the further development and validation of SCANAIR with the emphasis on power reactor applications.

In the future, after the feedback from new experiments, the collaboration with IPSN will continue with refined calculations of pellet-cladding interaction under RIA. The capabilities of calculating fuel rod behaviour in accident conditions will be further improved with the acquisition of FRAPTRAN code (new FRAP-T6 version) through NRC. The latter involves validation of the new parts of the code that have been renewed or are under development.

3.1.2 Modelling of fuel performance under RIA

A multidimensional mechanical approach (EPFMD) is applied to see if it would improve the predictions of fuel performance under RIA. Secondly, the significance of the fuel and cladding creep under RIA will be investigated, and if necessary, creep model should be added to the SCANAIR code.

Calculations on the pellet deformation kinetics in a French RIA-test (Rep-Na2) has been done by using VTT Energy's multidimensional mechanics code EPFMD (developed by Kari Ikonen). Several simplifications were made in this first study: Gap conductivity and other thermal properties were kept constant and plastic deformations were omitted. Enthalpy deposition was 210 cal/g and the pulse half width 9.5 ms. The temperature distributions in the fuel were produced by a simple thermal calculation at certain intervals. Due to the high plutonium-content in the outermost region of the high burnup fuel pellet, the power and temperature maximums are reached near the surface. This yields to thermal expansion, which is largest at the mid-elevation of the pellet, and the pellet takes a barrel shape. Heat flux is both inwards and out of the fuel rod. After 100 ms the temperature starts to level out within the pellet and after 1.4 seconds a typical parabolic distribution is seen. Still, the centre temperature is very high. The pellet then takes an hour-glass form and the pellet ends are in hard contact with the cladding. Fig. 5 displays the change of the pellet as calculated by the EPFM2D-code.

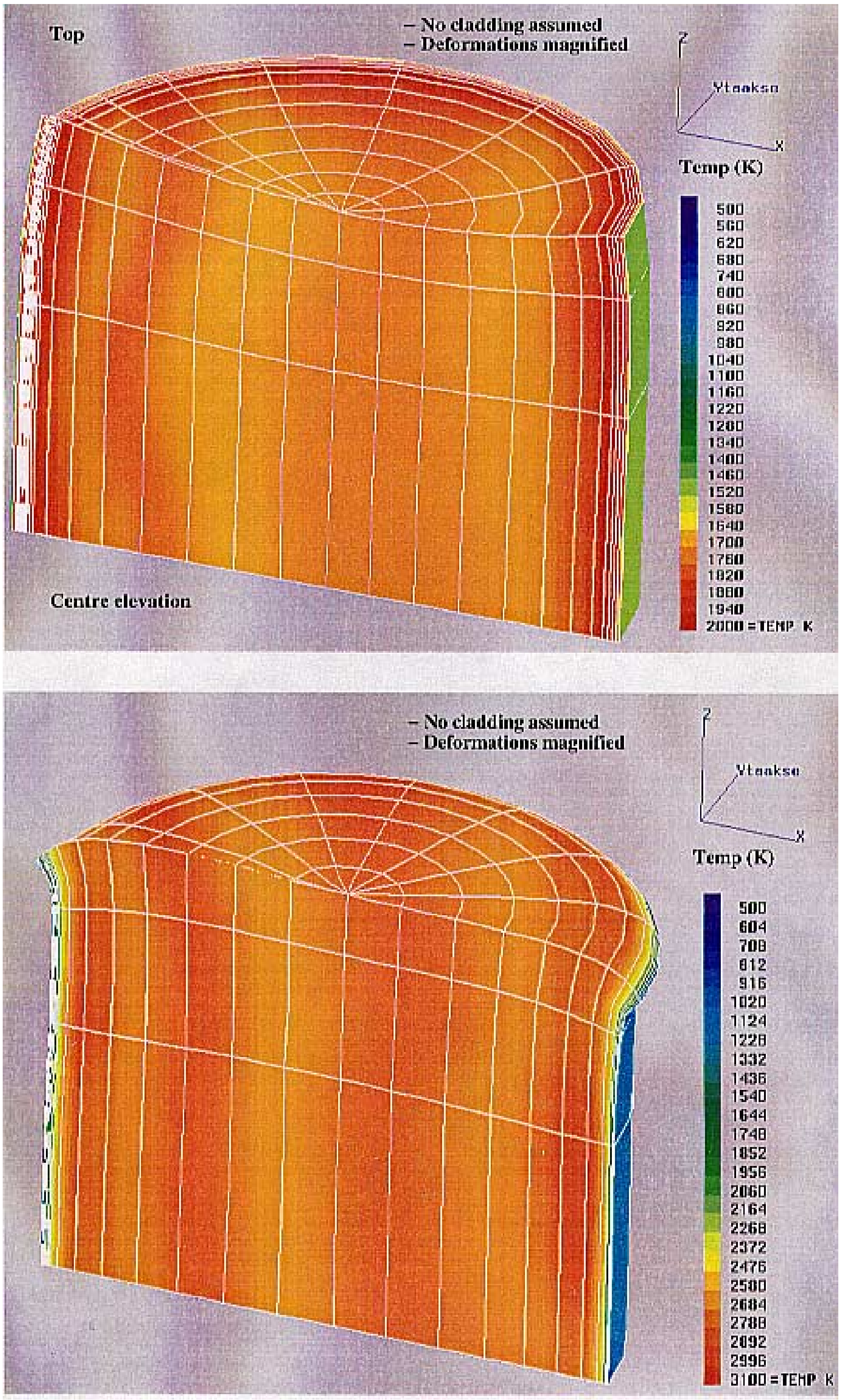


Figure 5. The pellet shape changes and temperatures at 20 ms and 100 ms after the initiation of an energetic pulse (RIA). Calculated with the EPFMD code.

The applicability of the multidimensional programs EPFMD has lately been improved (Ikonen 1996). Besides thermoelastic strains and plastic deformations, the creep component has been added to time dependent applications. The UO_2 and cladding creep do not only depend on the influencing stress but on temperature and neutron fluence, and equations for those were implemented in the code. A test case (14.5 MPa, 370 ^oC) showed encouraging results for long term cladding creep in comparison with measurement. For extreme conditions like those in RIA, the currently available material creep data and correlations are insufficient.

The higher the fuel burnup the larger the expansion of the outer rim region will be. In the future, the EPFMD program, complemented with a model for gas swelling from SCANAIR and with plastic deformations of fuel and clad, will be applied to Rep-Na2 test rod.

3.1.3 Validation of fuel behaviour models (ENIGMA)

The current version of the ENIGMA code is 5.9b. The code has been actively applied for analyses of the Finnish reactor fuel. Since the development of the code, there has been a world-wide tendency to go to increased fuel burnups, which places new requirements for the codes. For example, the code should be able to predict the initial conditions of a simulated accident with good accuracy.

FUMEX benchmark exercise

Good impression on the code’s capabilities has been confirmed by the international code benchmarking programme FUMEX organised by IAEA and conducted in 1993 to 1996 (Wiesenack 1996). Fifteen countries and 19 codes took part in the programme. Data on power histories, in-pile measurements and post irradiation experiments were provided by the Halden Project from 10 fuel rods. In the second phase of the programme, a comparison of the calculations were carried out after code improvements, followed by a sensitivity study to assess the codes’ response to changes of single parameters. The exercise was useful in demonstrating the strong points of the codes as well as pointing out where improvements should be directed.

Partly as a result of the FUMEX exercise, all of these codes now include modelling of fuel thermal conductivity decrease with burnup. It is this area of thermal performance where the greatest improvements have been made. Due to the strong temperature dependence of fission gas release, difficulties remain with predicting the fission gas release correctly. Major lack of progress was, however, found in the area of mechanical interaction. This is considered to be an important drawback with its implications in many other areas of fuel modelling.

Thermal performance of the ENIGMA code appeared to be good in conditions relevant in power reactors. In high power ramps and for xenon filled rods, which represent very demanding conditions, deviations in temperature and fission gas release do occur. As with most of the codes the mechanical interaction could not be estimated well enough, which was generally seen as an under prediction of rod elongation.

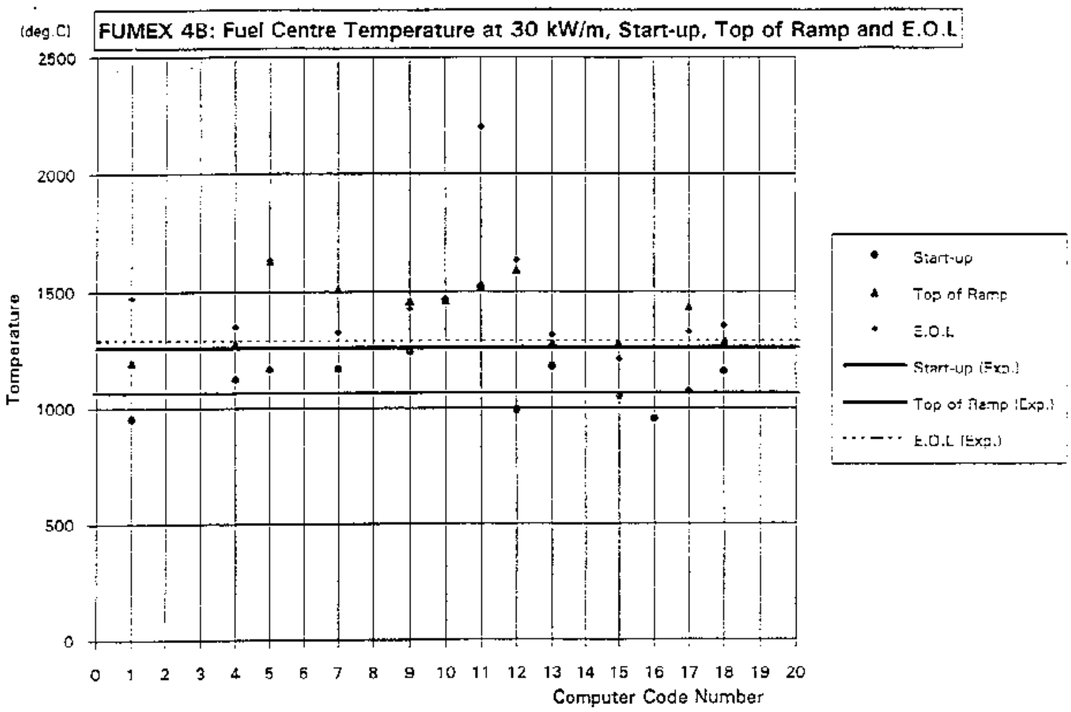
Fig. 6 displays a comparison of the calculated and measured temperature in one FUMEX case on three occasions: at start-up, during a power ramp and at the end of irradiation (VTT's ENIGMA is code number 4, and the two English ENIGMA code versions are marked with 17 and 18). Due to the high rating and low fill gas (He) pressure, there is early gas release followed by a burst release during the ramp at rod burnup of 20 MWd/kgUO$_2$. The scatter in the predictions of the codes is wide and often the post ramp temperatures are over predicted.

High burnup phenomena

Two segments, cut from a standard BWR fuel rod, base irradiated in a commercial reactor to a burnup of 59 MWd/kgUO$_2$, and equipped with re-instrumented thermocouples, were irradiated in the Halden reactor in IFA-597.2. The burnup is high enough for the changes in the UO$_2$ basic structure at the fuel pellet periphery (rim region). The effects of the porous rim on the temperature and its radial distribution were analysed by comparing the measured temperature with those of fuel rods in which such a rim was not supposed to exist (Ranta-Puska 1996a). The work was part of the co-operation between Halden Project and VTT.

When compared with the measured steady state temperature data and transient time constants from this burnup range, it was concluded that the fuel conductivity degradation of ENIGMA model is too strong. By using the recent fuel conductivity model suggested by the Halden Project, instead, the measured temperatures can be correctly reproduced. The thermal implication of a 200 μm porous rim is detectable, yet not dramatic, as shown in Fig. 7. At moderate powers, the radial temperature is somewhat distorted at the periphery. A small gap of 50 μm contaminated with fission gases would yield a similar profile. At 250 W/cm, the centre temperature will increase by about 100 ^{o}C due to low thermal conductivity of the rim. With increasing power, the influence of the gap will become weaker but the thermal resistance of the rim persists. Thus, with increasing power the temperature gradient across the fuel pellet radius becomes steeper, and the stored energy will be higher than in the no-rim fuel. In a power reactor, however, it is unlikely that fuel rods with over 50 MWd/kgUO$_2$ burnups will run at ratings exceeding 250 W/cm.

Validation and updates for even higher burnups with the aid of the results from international test programmes continues to be a necessity.



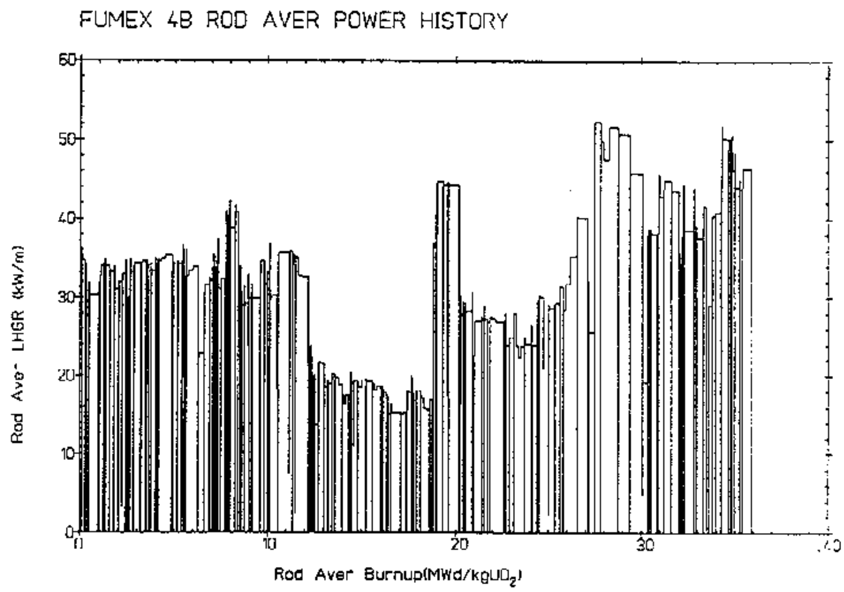


Figure 6. Fuel behaviour code predictions compared to the measured temperature for a FUMEX case (upper plot). VTT's ENIGMA is marked with code number 4. The rod has experienced moderate or high power through its irradiation (lower plot).

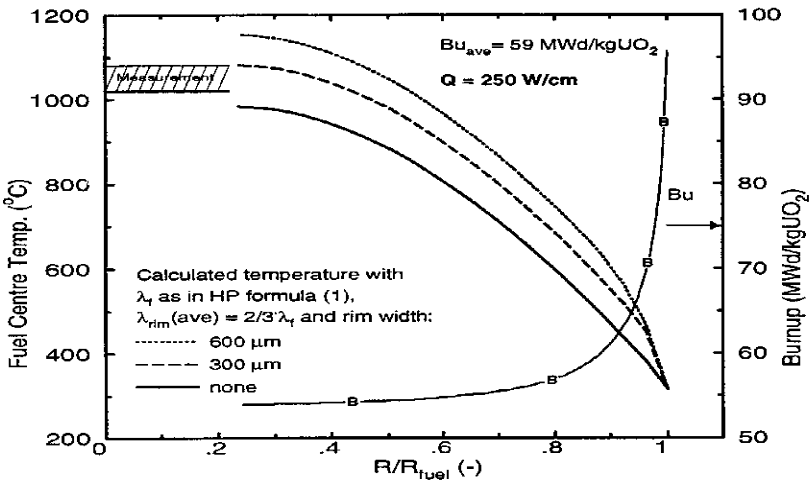


Figure 7. Simulation of temperature profiles with various rim widths. True rim layer thickness is about 200 µm and the centre temperature 1020 - 1080 °C.

3.1.4 VVER fuel rod irradiations

SOFIT programme

The objective of the SOFIT programme was to investigate the VVER-440 fuel behaviour in normal operational conditions. The experimental part of the SOFIT-program consisted of irradiation of clusters of pre-characterised and instrumented VVER-440 test rods in the MR reactor of the Kurchatov Institute of Russia. The first phase of the program (SOFIT-1) has been described in more detail in the final report of YKÄ Research Programme. In the second phase of the programme (SOFIT-2) it was planned to continue the investigations with emphasis on the mid- and high burnup range, whereas SOFIT-1 concentrated on fresh fuel and low burnup fuel.

Due to the permanent shut-down of MR reactor and delays in finding an adequate alternative experimental facility for these investigations, the SOFIT program was terminated at the end of 1995. Part of the in-pile data and PIE data of SOFIT-1 were released and submitted to an international data base of NEA (Turnbull 1995).

PWR/VVER experiment at Halden, fuel densification

A test irradiation of 12 VVER and PWR fuel rods started in 1995 in the Halden reactor. The rods are placed in two clusters with VVER and PWR fuel rods arranged in pairs having the same gap size and instrumentation. All cladding tubes are of the Russian type and manufacture, containing 1 % Nb in zirconium. The current test will significantly add to the measured temperature data for the VVER fuel, and will challenge the fuel behaviour codes that have been mainly developed and validated for western fuel types.

The temperatures are measured with expansion thermometers that are placed in the centre hole of the fuel stack and extend over the full length of the rods. In the axially averaged temperatures thus obtained, only marginal differences between the fuel types have been detected. From the elongation measurements, estimation of the fuel densification can already be made. The VVER fuel stack shrinks more than the PWR fuel despite the low initial porosity (3.0 %), Fig. 8. From the reduction in stack length, the volumetric densification was estimated at (Ranta-Puska 1996b):

	PWR fuel	VVER fuel
Densification	0.2...0.4 %	0.9...1.2 %

Well in line with the above VVER fuel in-pile densification, thermal resintering of 1.1 to 1.8 % has been reported in out-of-pile tests. Detailed information on the fuel microstructure, on pore and grain size distributions especially, is needed to explain the measured fuel stack length changes. The test will be continued up to rod burnup of about 20 MWd/kgUO_2.

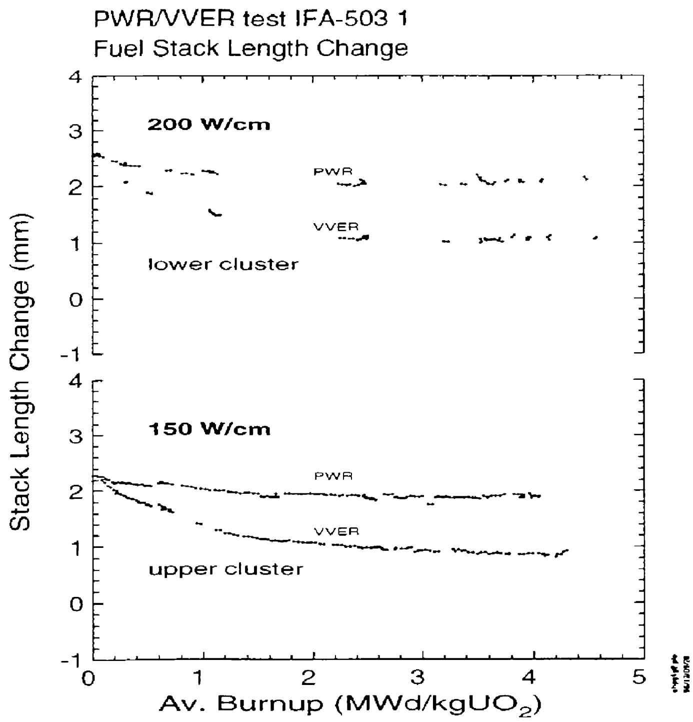


Figure 8. UO_2 stack shrinkage as a result of early sintering in the VVER/PWR comparative test irradiation at Halden.

3.1.5 Performance of cladding materials

Two of the international projects focused on the performance of cladding materials in undesirable conditions. The main emphasis was in studying the phenomena that potentially become management limitations for the light water fuel designs: rod over pressure and secondary defect.

Effect of rod over pressure

The accumulated release of fission gases may at high burnup cause the internal pressure to exceed the system pressure, and a re-opening of the gap resulting in deteriorated heat transfer, i.e. the phenomenon known as “lift-off“, can be postulated. Licensing of higher burnups may increasingly need to rely on the “non-lift-off“ rather than the customary “non-over pressure“ criterion. The kinetics of the creep deformation, the possible differences between the stresses either being compressive or tensile, and the rate of the UO_2 swelling govern the re-opening of the pellet-to-clad gap.

In two Studsvik projects, ROPE I and ROPE II, the problem was experimentally approached for BWR and PWR fuel, respectively. The results of the first project confirmed that BWR cladding can withstand a certain level of over pressure without lift-off for more than two months. The results of ROPE II show a similar behaviour of PWR fuel (Stengård 1996). Six fuel rods from two fuel suppliers, base irradiated to high burnups in commercial reactors, were tested in conditions typical of a PWR. Two test assemblies, one of each type of the fuel, had one high pressure rod (13 MPa hot over pressure), one rod with low over pressure (7 MPa), and one reference rod without over pressure. The diameter increase of the high pressure rods was 23 and 17 µm (A1, K1, Fig. 9), and of the low pressure rods about 8 µm, during the irradiation of 110 full power days. The slight differences in diameter changes between the rods with the same over pressure (A vs. K) may be due to different clad cold work and chemical composition. Although the clad creepout exceeded the normal pellet swelling, there was no measurable decrease in the gap conductance in the over pressure rods. The ROPE II project demonstrated that a PWR fuel rod with an appreciable internal over pressure can be operated at about 200 W/cm for more than 3.5 months without adverse effects.

The ENIGMA code calculations showed satisfactory agreement with the data as regards the deformation of the ROPE II rods.

Secondary defect behaviour of LWR fuel

In the international Studsvik DEFEX project (Defect Fuel Experiment) new technique was introduced to investigate the secondary defect behaviour of a commercial LWR fuel rod. The primary defect was simulated by a novel arrangement of steam injection from the top plenum region.

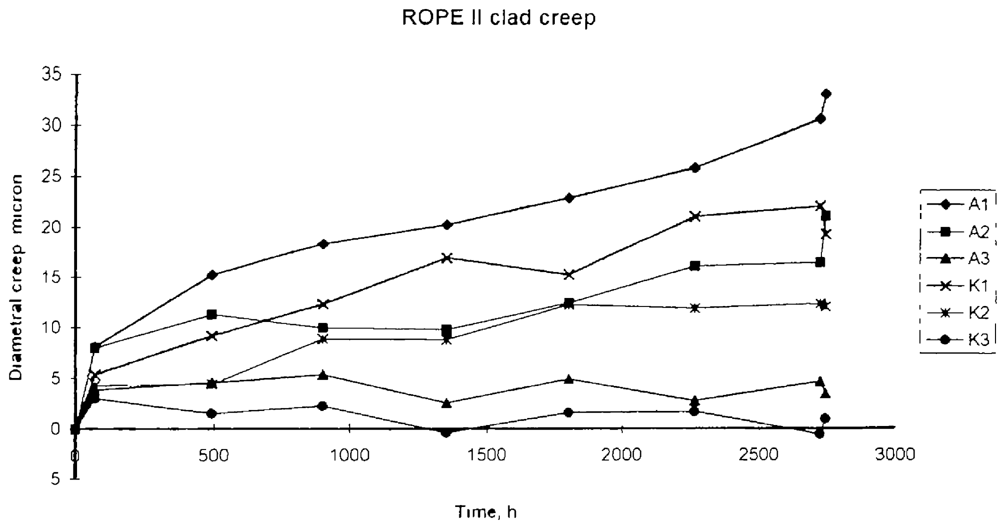


Figure 9. Rod diameter changes in the international rod over pressure experiment ROPE II. (A and K represent two different cladding types; in A1 and K1 had high over pressure, while A3 and K3 had no over pressure).

New results of the project confirm the earlier understanding of the hydriding scenario. Water penetrates through the primary defect, flashes into steam and fills the fuel cladding gap. The steam will oxidise the fuel and the internal surface of the cladding and therefore the amount of hydrogen will be increased and picked up by cladding. For reactor temperatures the following hydriding criterion for the partial pressures of steam and hydrogen has been proposed (Gräslund 1995):

$$P_{Steam}/P_{Hydrogen} < 0.01 \quad \text{[assuming } P_{hydrogen} > 20 \text{ mm Hg].}$$

When the fuel-to-clad gap is partly closed, as is the case with irradiated fuel at typical powers, diffusion of steam will be reduced and the above criterion may be fulfilled at some axial distance from the primary defect.

An oxide layer on the inner surface of the cladding will delay the initiation of hydriding. As the DEFEX-rods were all fresh, the clad inner surfaces were not oxidised during the tests. This caused accelerated reactions and quick degradation of the fuel rods. It was concluded that the tests are not sufficiently representative for the power reactor cases, where Zircaloy surfaces will be oxidised already at low burnup.

The experience has led to a proposal for the DEFEX II Project, where rods pre-irradiated to medium burnup would be tested with simulated defects. The planning of the new project is underway.

References for section 3.1

Gräslund, C. 1995. Final report of the DEFEX project. STUDSVIK-DEFEX-21, 1995.

Ikonen, K. 1996. Creep in thermo-plastic deformation and stress analysis. VTT Energy, VAHTI-8/96 & PATRA-3/96.

Kyrki-Rajamäki, R. 1995. On the role of burnup effects of fuel properties in RIA analyses. Paper presented at the specialist meeting on Transient behaviour of high burnup fuel, Cadarache, France, September 1995.

MOU 1995. Memorandum of understanding (MOU) between the Technical Research Centre of Finland and The Commissariat à l’Energie Atomique - Institut de Protection et de Sûreté Nucléaire. September, 1995.

OECD 1995. Scientific issues in fuel behaviour. Nuclear Energy Agency (NEA), OECD documents, January 1995.

Ranta-Puska, K. 1996a. Thermal effects of fuel conductivity degradation and RIM in IFA-597.2 (BWR) and IFA-533.2 (HBWR) rods. Paper presented at the enlarged Halden Programme Group Meeting (EHPG), Loen, Norway, May 1996.

Ranta-Puska, K. 1996b. IFA-503.1 (VVER/PWR Test); ENIGMA Calculations compared with the first measurement data. EHPG, Loen, Norway, May 1996.

Stengård, J.-O. 1996. Summary of ROPE II -project (in Finnish). PATRA-8/95.

Turnbull, J. A. 1995. Review of nuclear fuel experimental data. Nuclear Energy Agency (NEA), OECD documents, January 1995.

Wiesenack, W. 1996. Draft document IAEA CRP FUMEX. Version 4, 13-08-1996.

3.2 Reactor physics and dynamics

Code development in the fields of reactor physics and dynamics, as well as thermal hydraulics, has been one of the key areas of reactor safety research in Finland since the middle of the seventies. A comprehensive and independent reactor physics and dynamics code system has been created at VTT Energy for both BWR and VVER reactors. The code system is being modernized all the time both by own development work and by international cooperation. At present most of the licencing analyses of the Finnish nuclear power plants can be performed with our own codes. The code system has been widely used by the nuclear safety authorities and by the Finnish nuclear power companies as well as by customers abroad.

The scope of the DYNAMIC project is reactor physics and reactor dynamics with related thermal hydraulics, which are essential in confirmation of the safety of nuclear power reactors. The main objective of the project is to upgrade and extend the reactor physics and dynamics code system for tasks that are related to increased requirements and recent safety concerns. Applicability of the reactor physics codes needs to be ensured also for high burnup fuel. New methods have to be developed for the production of reliable and validated input data for reactor dynamics codes and other nuclear applications. In reactor dynamics the objectives are to complement and validate the calculation system for complex reactivity accidents, such as ATWS, boron dilution and BWR core stability.

Introduction of new fuel types in connection with the ongoing modernization projects is considered at both Finnish NPPs, and very extensive studies have been carried out needing the expertise of the whole project group: fuel management and related reactor physics data, reactor dynamic transient analyses, and thermal hydraulics in the bundles. Studies have been carried out also for foreign customers.

3.2.1 VTT’s calculation system for reactor physics and dynamics

VTT Energy’s calculation systems for reactor physics and dynamics is shown in Fig. 10. It consist of codes for both types of Finnish nuclear power plants: BWR and VVER reactors. Many computer codes have been acquired through the international data banks to complete VTT’s calculation system. As compensation, some Finnish codes have been delivered to the data banks. Some of the reactor physics programs have been acquired on a commercial basis.

In earlier projects, many indigenous computer codes have been developed mainly because Finland was the only western country using VVER-type reactors, which in many respects differ from typical PWRs (e.g. hexagonal fuel assemblies and horizontal steam generators). During the recents years the available resources have been devoted mainly to upgrading the existing calculation system.

REACTOR ANALYSIS
CALCULATION SYSTEM OF VTT ENERGY

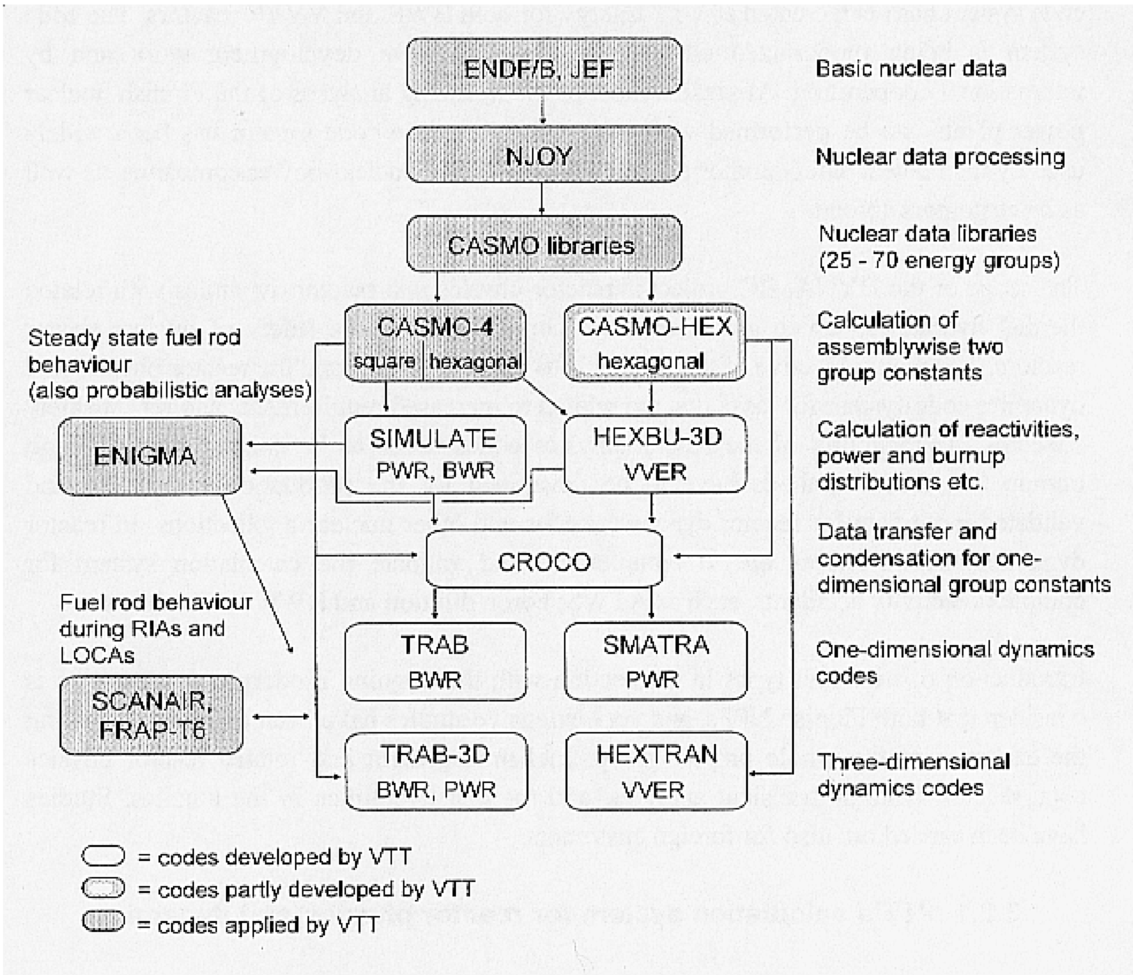


Figure 10. Calculation system of VTT for reactor physics and dynamics.

Recent updates in the system include the installation of many international codes, eg. the latest version of the NJOY processing code, the SCALE4.3 program package, the MCNP4A Monte Carlo code, the advanced three-dimensional two-group reactor analysis code SIMULATE-3 of Studsvik Core Analysis Ab, and the first production version of Studsvik's fuel assembly program CASMO-4 (for square, hexagonal and cluster geometries).

All the reactor dynamics codes have been developed in VTT and also the main targets of the new development work are at present the dynamics codes: three-dimensional modelling of the neutron kinetics also in rectangular geometry and renewing the thermal hydraulic solution methods.

One of the goals of the project is to maintain Finland's international contacts in the field of reactor physics (OECD/NEA Nuclear Science Committee, IAEA). The project contributes to the improvement of VVER safety by participating in the AER, the International Cooperation on VVER Physics and Dynamics. AER is an independent organization started in 1991 and open to all interested parties. At present 22 research institutes or power companies from 8 countries (Bulgaria, Czech Republic, Finland, Germany, Hungary, Poland, Russia and Slovakia) have joined in the cooperation. VTT participates in the AER working group on VVER reactor safety analysis, and in the new working group on spent fuel problems, radwaste and decommissioning of NPPs. In 1996 VTT organized the annual symposium of AER in Finland (Vidovszky 1996).

3.2.2 SFAV six-equation model for two-phase flow

A new formalism for two-phase flow has been derived: Separation of the Flow According to Velocity (Rajamäki & Saarinen 1994b, Narumo & Rajamäki 1995, Rajamäki & Narumo 1995, Narumo & Rajamäki 1996a, 1996b, Narumo 1997). SFAV two-phase flow model consists of six conservation equations for the mass, momentum and energy of two fluids like any conventional two-phase flow model. The difference has been made with the derivation of the momentum equations: the flow is separated into two velocity fields and conservation equations are derived over their domains. The basis is clearly physical: mass and energy are distributed approximately according to the phases, but the variations of momentum follow more the flow velocity.

In the SFAV-formalism the distributions of the phases and their velocities are not treated as uniform in the cross-sectional areas occupied by the fluids, but transverse spatial dependencies are allowed. With suitable dependencies and without using nonphysical fittings, the equation system is well-posed; in other words, the characteristic velocities are real. The couplings to unknown variables create obstacles to numerical solution, but they are easily overcome with the new shape-preserving solution method PLIM. The SFAV model has been shown to give sonic velocities in very good agreement with measurements over a full range of void fractions. The calculated propagating velocities of void fraction disturbances agree well with measurements. The work of T. Narumo on the SFAV model has been funded with a personal grant from the Academy of Finland.

The characteristic velocities of the SFAV-equation system can be associated with the propagation velocities of small disturbances. They have been shown to be real in any flow conditions. This does not prevent a model to predict physical instabilities in case needed.

In addition, the characteristics have been verified to be very well in agreement with measurement data obtained at low pressures (Fig. 11 and 12); relevant data at high pressure has not been available yet.

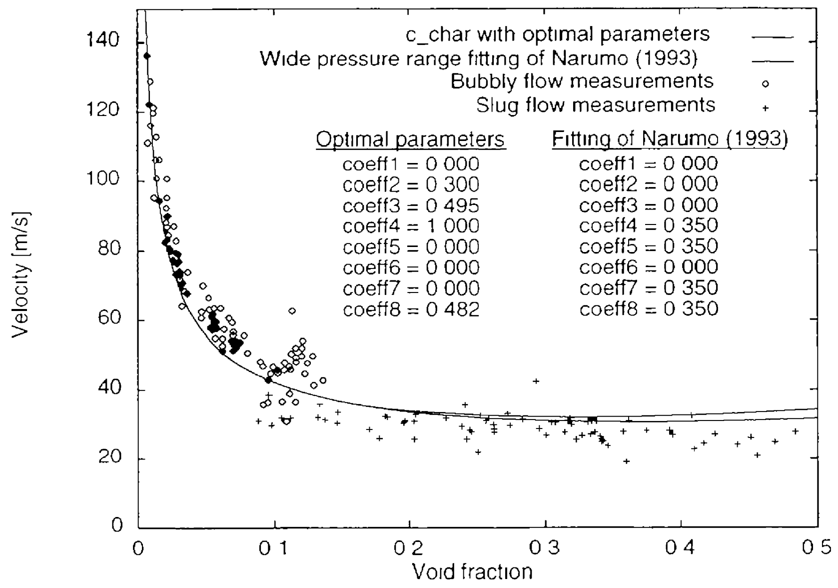


Figure 11. The characteristic sound speed from the SFAV-model compared with measurements on pressure wave propagation in bubbly and slug flow regimes with general purpose SFAV-parameter values and optimal parameter values for this case.

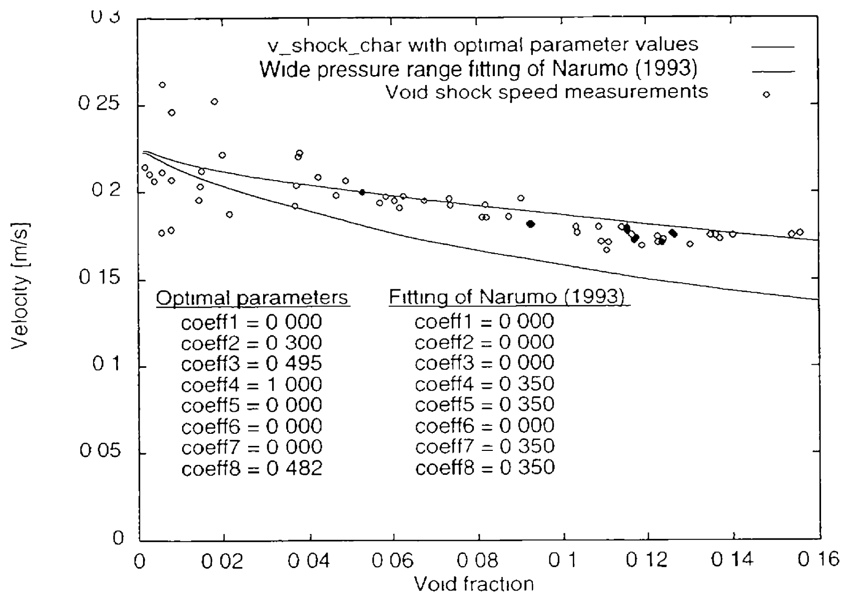


Figure 12. The void fraction shock speed from the SFAV-model compared with measurements with general purpose SFAV-parameter values and optimal parameter values for this case.

The separation of the flow into two velocity fields unavoidably leads to a bigger number of closure relations than in the case of the conventional model in which the flow is separated into subflows according to the phases.

Fortunately, the best way for separating the flow into such two subflows that the distribution of mass and energy is described as well as possible, is to perform the separation according to the phases like in the conventional model. That is why most of the source terms in the SFAV-model can be obtained directly from the existing two-phase flow models.

The most important sources that have to be modified for the SFAV-model are the wall friction terms and the term for friction between the velocity fields (the interfacial friction term). The wall friction is modelled with two terms describing the effect of each velocity field (or SFAV-subflow) on the wall separately. The advantage is that more detailed effects can be described fairly easily, e.g. when major part of bubbles occupy the neighbourhood of the channel wall in the low void fraction case. This approach requires wall friction factors for both flows. In addition, the interfacial friction factor is needed. All of these three factors can be approximated by using known correlations for two-phase flow wall friction and velocity coupling (drift-flux and slip-correlations). Advantage can be taken of the momentum equation that mostly determines the velocity difference of the subflows. In steady-state conditions, that equation should contain the same information as a suitably chosen drift-flux correlation. Another condition is obtained from existing wall friction correlations, namely the sum of the wall frictions of the SFAV-subflows should equal the total wall friction predicted by such a correlation. The friction factors can be solved in one go by minimizing a suitably chosen and weighted error functional, which in the ideal case leads to the friction factors that agree with both of the correlations.

The friction factors calculated by this means are, in general, not too smooth (Fig. 13). This is due to the contradictory nature of the correlations that the factors are based on. However, some smoothing of the factors for numerical calculations can be easily justified because the factors only set the scale of the friction terms but do not determine their dynamic effects.

There are still gaps in the friction factors because of the limited validity range of the used drift-flux correlations and wall friction correlations. The possible means to fill the gaps are the interpolation over them, which may be vague, or using experimental data, which are difficult to find. Probably the task will be completed by using a combination of those.

A SFAV-model containing the most essential features as concerns flow dynamics, has been programmed into the CFDPLIM-hydraulics solver. Basic validation for the derived friction model was made by searching steady-state distributions of the system resulting from different initial conditions and comparing to the steady-state given by a drift-flux correlation. The calculated cases were very well in agreement with each other.

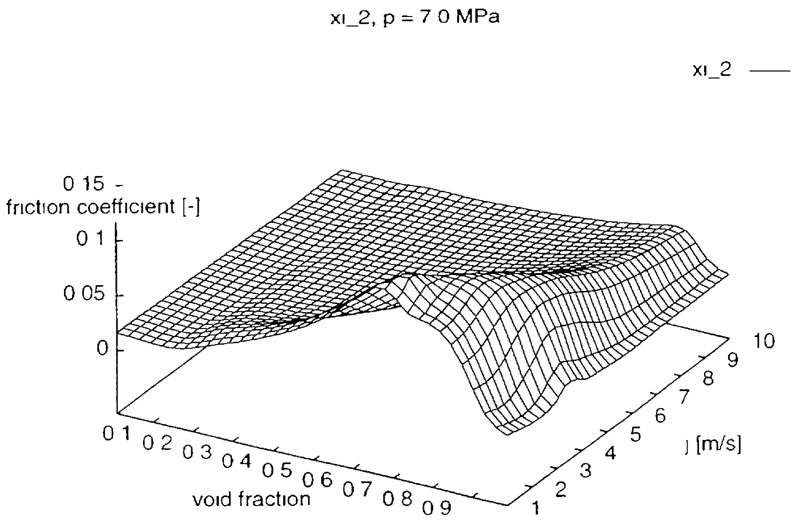


Figure 13. Example of calculated wall friction factor for SFAV-model; friction factor for the heavier subflow.

As an application, response of the flow in a partial BWR fuel channel has been calculated as function of frequency of a sine-wave inlet disturbance. The results have been compared to those obtained with the corresponding drift-flux model. This kind of situation could happen in practise e.g. with feedwater pump partial damage or blockage in the fuel channel. The results obtained by each model differed from each other already with circa 5 Hz frequency (Fig. 14), while the time constant for the relaxation of steady-state distributions would have implied that significant differences do not appear until with frequencies of order 50 Hz. This may be an evidence that accurate modelling of flow dynamics is necessary also for fairly slow transients.

The validation and testing of SFAV will continue with applications to critical flows, counter-current flows, horizontal flows and flows with greatly varying void fraction. The long-term goal is to apply SFAV in VTT's reactor dynamics codes.

3.2.3 Computational Fluid Dynamics code on the basis of PLIM method (CFDPLIM)

PLIM, Piecewise Linear Interpolation Method, is a new highly accurate shape-preserving characteristics method for solving systems of one-dimensional hyperbolic partial differential equations (Rajamäki & Saarinen 1991, 1994a, Rajamäki 1996a, 1996b). PLIM is applicable and accurate always when conventional methods are accurate and is able to

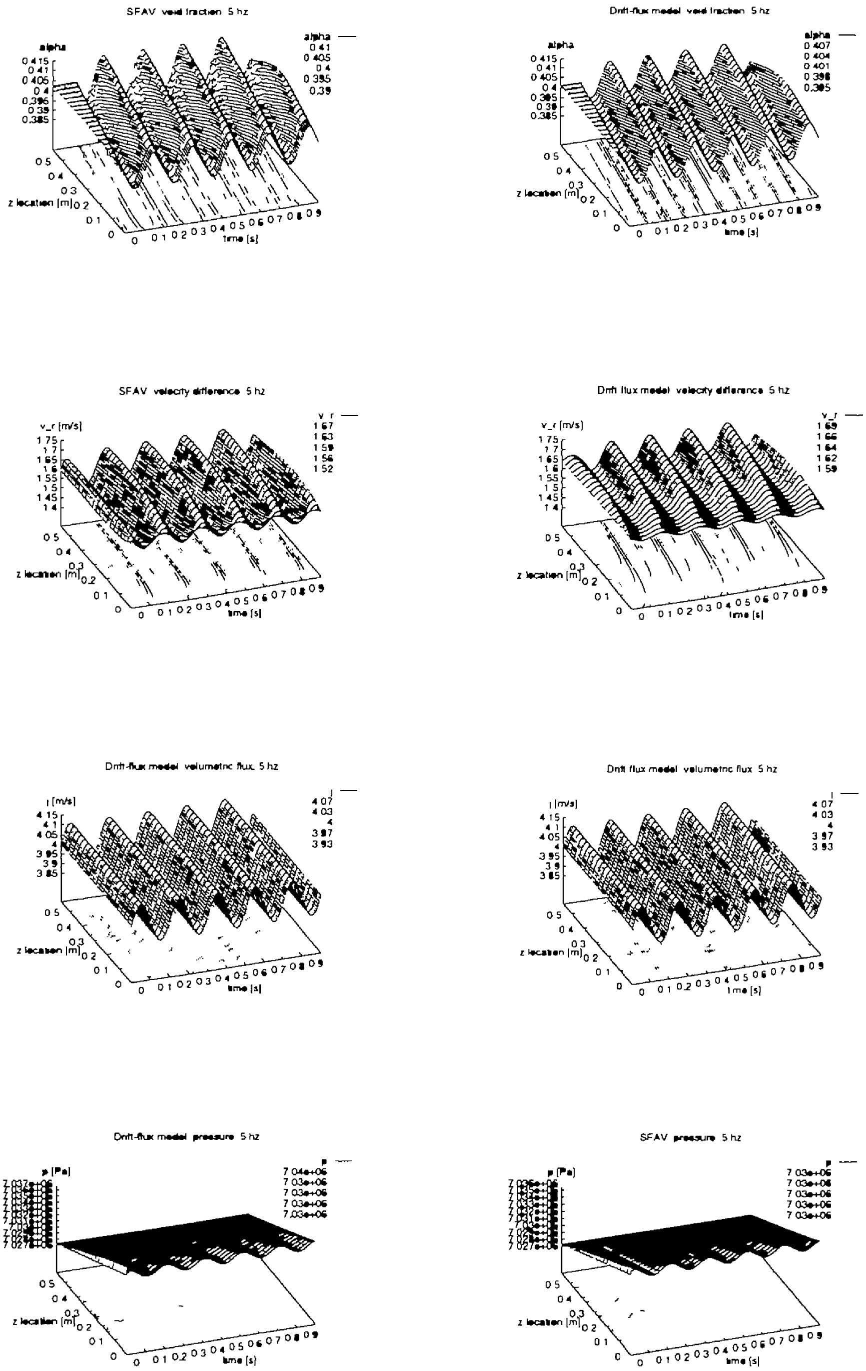


Figure 14. Response of the SFAV-model and the drift-flux model on the 5 Hz sine-wave disturbance in inlet water volumetric flux.

treat propagating piecewise linear distributions accurately on a mesh grid. In the one-dimensional time-dependent case, interpolation with the piecewise linear polynomial approximation containing two unknown parameters yields the desired shape preserving scheme. The conservation laws are not violated either. The discretization mesh needed and the numerical performance of the solution are in direct proportion to the physical complexity.

PLIM method has been successfully tested in several demanding flow problems, e.g. stratified two phase flow, gas dynamics and various convection diffusion problems (Saarinen 1994). The numerical solution can handle all cases of reversed flow. Strong interactions due to source terms of the flow equations are allowed and movable discontinuities such as water levels can appear or disappear.

Development and testing of the Computational Fluid Dynamics code CFDPLIM based on the PLIM method has continued aiming at the reactor dynamics applications of CFDPLIM. The CFDPLIM code has proved to be easily transportable, and applications are presently running on different workstations and PCs.

CFDPLIM solves a general multi-variable system of flow equations. In order to be capable to do this, CFDPLIM presupposes that the terms of the equations can be locally linearized. We have encountered a special problem in representing of the phase change rate. When a phase appears or disappears, the strong coupling due to phase change alters discontinuously and the straight-forward linearization produces a too erroneous, even non-physical, approximation. A new methodology being capable to handle these discontinuities has been developed to representation of the phase change rate in CFDPLIM-functions.

In the reactor core it is a good approximation to suppose the minor phase of two-phase flow to be saturated. The minor phase in TRAB with the four equation system is always assumed to be steam. A form of the energy equation for CFDPLIM has been developed, which allows the minor phase transition from steam to water and, further, overheated steam.

CFDPLIM uses the same algorithms for all flow channels. The movable discontinuity, as water level, is taken into account with two movable discretization points. There remain, however, problems in the description of how the discontinuities appear or disappear. Viz., when the discontinuity is growing, the local linearization of the terms of the flow equations in the mesh interval becomes a poor approximation. The first weapon of CFDPLIM is the movable locally denser discretization which decreases the errors and aids temporarily. However, because CFDPLIM solves the equations accurately, the developing discontinuity is before long localized in one mesh interval, in which the local linearization is again an invalid approximation. Then the solution of CFDPLIM does not converge, although the error in itself may be acceptable. For this reason the discontinuity behaviour must be approximated. It is forced to spread into several mesh intervals of CFDPLIM. Diffusion flattens any behavior and particularly effectively when the variations are abrupt. When desirable, CFDPLIM utilizes artificial diffusion with a diffusion coefficient being a function of steepness of the behavior. In order to be generally operable in these situations, the numerical modeling of diffusion has to be very advanced. A new diffusion model has been developed and tested. It has been applied in an example case to describe the appearance, movement and disappearance of water level (Rajamäki 1996b). The varied parameters have been flow velocity, cross sectional area, and evaporation rate. The tested cases simulate e.g. emptying of a channel due to boiling and emptying of a bottle.

Development and testing of CFDPLIM will be continued in the DYNAMIC project in order to improve modelling of features needed in PLIM-based reactor dynamics applications.

3.2.4 Application of PLIM in the reactor dynamics codes

Application of the new hydraulic solution method PLIM (Piecewise Linear Interpolation Method) in the reactor dynamics codes aims at improving the accuracy of the dynamics codes in challenging flow conditions, removing of modelling restrictions, and for better utilizing of modern computer capabilities. PLIM is also a prerequisite for applying the evolving new thermal hydraulic model SFAV in the dynamics codes in the future. The PLIM method totally eliminates numerical diffusion and dispersion, eg improving the tracking of boron and temperature fronts during transients. The application of PLIM is based on the CFDPLIM code.

Early testing of CFDPLIM in the dynamics codes was carried out in 1994 with the three-dimensional core model of HEXTRAN in one-phase flow in a multi-channel geometry. The new model proved to be useful in confirming the excess conservativity normally applied in HEXTRAN to counteract the deficiencies of traditional methods during a boron dilution transient in natural circulation conditions (Kyrki-Rajamäki & Stenius 1995). A simple description of a VVER circuit model based on CFDPLIM was also tested.

Further development and testing of CFDPLIM in the dynamics codes is presently carried out in the one dimensional TRAB core model. Most of the early coding has been revised and updated, and the modelling now comprises a full-coupled neutronic and two-phase thermal hydraulics calculation. Core channel geometry has been extended to include axial subchannels and discontinuities between them. User-friendliness of the code has been improved eg. by including a coupled restart and several graphics output options, and allowing user-defined features normally used in the production versions of TRAB.

Two-phase modelling utilizing the CFDPLIM code as the hydraulics solver in TRAB has been tested with both the hot channel model and the coupled neutronics, heat transfer and hydraulics calculation in a single TRAB core channel in both VVER and BWR conditions. The calculated transient cases have been the hot channel of the third AER benchmark, a VVER-440 control rod ejection from zero power conditions with full feedback effects and thermal hydraulics modelling, and NEANSC three-dimensional BWR core transient benchmarks, calculated in one dimension and slightly modified for testing purposes. The NEA benchmark cases are a core undercooling and a core pressurization transient, and they were used to test both the hot channel and the coupled core channel model.

Preliminary results have confirmed the applicability of the coupled and uncoupled two-phase calculation both in steady state and in dynamics. The model has proved to function as a whole, and the results for the calculated transients agree well with the results of the production version of TRAB (Fig. 15).

The goal of the TRAB development is a full reactor dynamics BWR circuit model utilizing CFDPLIM. Updating and testing of the earlier planned basic CFDPLIM model for a BWR circuit is planned for 1997. The development of a realistic VVER circuit model based on CFDPLIM for HEXTRAN will also be started in 1997. Further development of the core model aims at both streamlining the model and improving its user-friendliness.

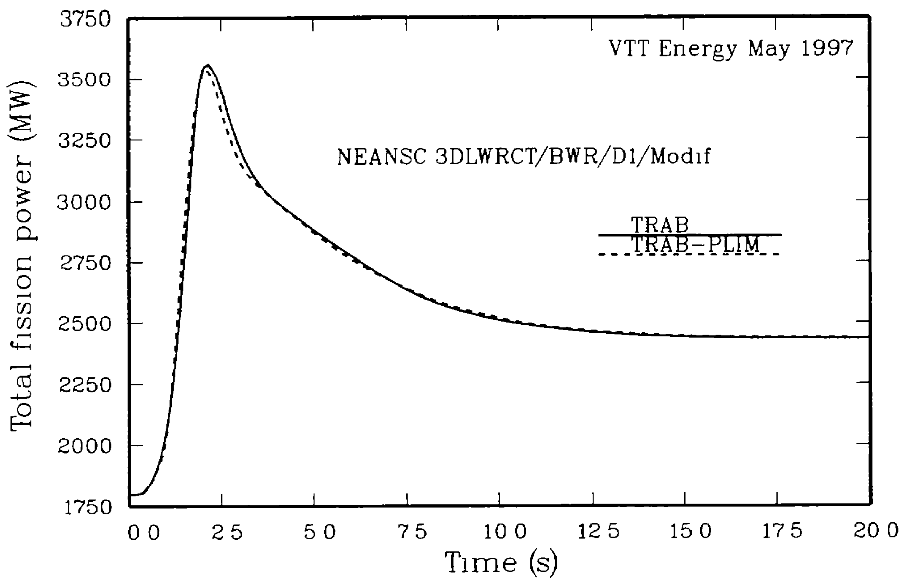


Figure 15. Comparison of results by TRAB and TRAB-PLIM for modified NEANSC 3D BWR core undercooling benchmark.

3.2.5 Three-dimensional reactor dynamics methods in safety analyses

A doctoral thesis prepared in the project on development, validation and application of the three-dimensional hexagonal reactor dynamics code HEXTRAN was accepted in 1995 (Kyrki-Rajamäki 1995b). As the result of the thesis it was shown that three-dimensional time-dependent power distributions of the reactor core can be calculated with such accuracy and versatility that they can be utilized in real analysis work. The calculation models of HEXTRAN are so effective that all types of licencing analyses in which the fission power generation behaviour is important can be carried out in full three dimensional core geometry. The physical time of the calculated cases has ranged from a few seconds to hours.

HEXTRAN models accurately the VVER core with hexagonal fuel assemblies. The code uses advanced mathematical methods in spatial and time discretization of neutronics, heat transfer and two-phase flow hydraulics. The dynamic coupling with the thermal hydraulic system code SMABRE allows also the modelling of cooling circuits.

Best-estimate or conservative analyses can be performed for different accidents, e.g. RIA, ATWS or local boron dilutions. The usefulness of the three-dimensionality is shown particularly when there are asymmetric changes in the fission power distribution originating from local neutronic or thermal hydraulic disturbances in the core or cooling circuits. HEXTRAN has been extensively applied for transient and accident analyses of VVER type nuclear reactors in Finland and abroad; analyses have been made for Loviisa and Paks NPPs and for the new Russian plant concept VVER-91.

Studies on the effect of high burnup during reactivity transients

In the NEA/CSNI Specialist Meeting on Transient Behaviour of High Burnup Fuel there was a session on Reactivity Initiated Accident (RIA) calculations; most of the sessions concentrated on fuel behaviour issues. Ten papers were given on Design Basis Accident (DBA) RIAs concerning results of high burnup fuel in realistic cores, including VTT's paper "On the role of burnup effects of fuel properties in RIA analyses" (Kyrki-Rajamäki 1995a). However, models for burnup dependent properties of fuel were not included in any other reactor dynamics code than HEXTRAN.

In HEXTRAN calculations, slow and fast RIA analyses of control rod group withdrawal and control rod ejection accident, respectively, were used as examples analyzing the burnup effects of fuel properties. They were carried out for a VVER-440 type reactor in typical EOC conditions. The consequences of the degradation of the fuel pellet conductivity with increasing burnup were studied. Comparison calculations were made using two different sets of input data for the conductivity: curves for fresh fuel without any burnup dependence and curves with terms depending on the actual burnup of each calculational node. The data of the UO_2 conductivity degradation given by Vitanza was used (Vitanza,1991): e.g., the fuel conductivity at 727 °C deteriorates over 20 % from its original value with modest assembly average burnups of 40 MWd/kgU.

The consequences of the degradation of the fuel pellet conductivity with increasing burnup were studied. The results showed that there was considerable increase in fuel temperatures of assemblies with higher burnup when the fuel conductivity degradation was taken into account (Fig. 16). However, no hot channel calculations were made, so the maximum temperatures in Fig. 16 are moderately low. Because the analyses were carried out with the three-dimensional HEXTRAN code, it can also be concluded that the Doppler feedback effect cannot fully compensate for the temperature increases by decreasing the fission power levels of hot assemblies. Also the effect of the burnup- and temperature-dependent gas gap conductivity was taken into account in the calculations. The fuel temperature increase was seen in slow and fast RIAs and in the stationary states at nominal power level. The effect is stronger in slow transients initiating from nominal power level.

The conclusion based on the HEXTRAN analyses is that it cannot be assumed in the RIA calculations that all the highest temperatures or enthalpies always occur in the fresh fuel assemblies, i.e., in the assemblies with highest power density. High temperatures can also be found in assemblies with high burnup if they are loaded in the middle part of the reactor core or near the control rods. The effect must be considered when the maximum enthalpy criteria or core loading types are planned. In the conclusions of the CSNI meeting it was mentioned that the conservativity of most calculations was not sure due to the lack of the burnup dependence of the fuel properties.

3.2.6 Development of the three-dimensional BWR dynamics code TRAB-3D

TRAB-3D is an adaptation of the three-dimensional dynamic code HEXTRAN to rectangular fuel assembly geometry (Kaloinen & Kyrki-Rajamäki 1997). The main difference between the codes is the solution of the neutronic equations since the neutron kinetics and one-dimensional thermohydraulic solution methods do not depend on the core geometry. A new method to solve the nodal diffusion equations in rectangular geometry was designed and programmed for TRAB-3D. The rectangular solution method of TRAB-3D is based on the nodal flux model of the HEXBU-3D and HEXTRAN codes and its accuracy is comparable to that of the hexagonal method. The following features are common for both methods:

- two neutron energy groups
- group fluxes are constructed from two spatial modes
- fundamental mode is approximated by polynomials and transient mode by exponential functionscontinuity of face-averaged flux and current at nodal interfaces
- fast two level iteration method to solve the nodal equations
- discontinuous flux at transverse interfaces of nodes

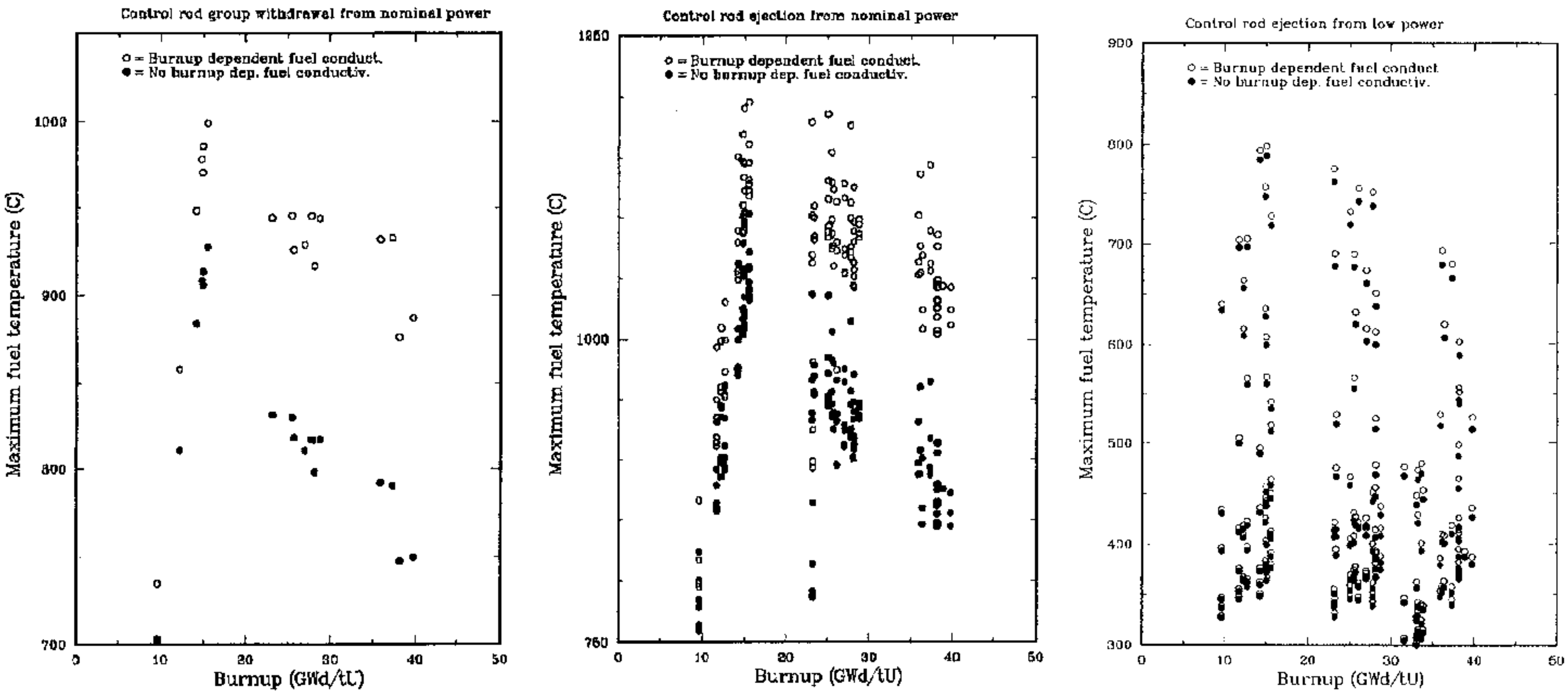


Figure 16. Maximum fuel center temperatures of fuel assemblies calculated with two types of data: with burnup dependent fuel conductivity or with fresh fuel data. From left to right: at the time of the maximum power during control rod withdrawal transient case, the control rod ejection case from nominal initial power level, and the control rod ejection case from low initial power level.

The transverse approximations of the nodal flux have two degrees of freedom per nodal face in each neutron energy group. Therefore, in addition to the continuity of average flux and current, the continuity of their moments are required at nodal interfaces. This leads to calculation of two sets of coupling coefficients between neighboring nodes. In hexagonal geometry the number of nodal faces is larger and more degrees of freedom can be satisfied without additional continuity conditions.

The rectangular solution method has been validated against accurate fine-mesh finite-difference calculations. Since in the axial direction the nodal model of TRAB-3D is similar to the model in HEXTRAN the calculation were made in two-dimensional geometry. Fig. 17 gives a comparison of assembly powers in a reactor core which consists of homogenized fuel assemblies and inserted control rods. The reference solution was calculated with the finite-difference code GOG using a small mesh spacing of 0.5 cm. The comparison shows that TRAB-3D solves the two-group diffusion equations with an accuracy of better than 1 % in assembly powers. Somewhat unexpectedly the accuracy does not seem to improve when the number of nodes in the calculation is increased from one to four nodes per fuel assembly.

				15 311 2 2
			13 982 0 2	14 834 - 1 - 2
		10 1 215 0 2	11 1 169 0 - 1	12 989 0 1
	6 1 189 0 2	7 1 284 0 - 1	8 1 083 0 2	9 921 - 2 - 2
1 625 5 5	2 1 141 - 1 - 4	3 1 200 0 1	4 1 040 - 1 - 3	5 458 4 3

625	GOG, reference,	k_{eff} = 1 06595
5	TRAB-3D, 1 node/ass ,	k_{eff} = 1 06575
5	TRAB-3D, 4 node/ass ,	k_{eff} = 1 06572

Figure 17. Deviations (%) of assembly powers calculated with TRAB-3D from the reference results in a 45-degree symmetry sector of the test reactor.

TRAB-3D has a detailed description of fuel and control assemblies for both BWR and PWR reactors. A fuel assembly can consists of several axial segments of different composition (enrichment, burnable absorbers, etc.) and each segment may be associated with a different set of homogenized cross sections depending on the position of control rods. According to the reactor type the control rods are inserted from bottom or top of the core and also part length rods can be treated in the code. To account for the void history effect which has a major importance in BWR calculations the input cross sections are given for different values of historical densities (void fraction) of the moderator. The nodal distributions of the densities including the possible effect of control rod history are obtained from fuel management calculations with, e.g. the POLCA code.

A three-dimensional test calculation with TRAB-3D for a TVO core has given results which are in a very good agreement with the results of the POLCA4 code. Fig. 18 shows a comparison of assembly powers calculated with the codes for the core at full power. TRAB-3D gives a slightly higher (+240 pcm) value for the effective multiplication factor than POLCA4. The average deviation in assembly powers between the calculations is less than 1 % and the maximum deviation is less than 3 %. The radial and nodal peaking factors agree within 0.5 % and 5 %, respectively. Since the latter factor appears near the tip of an inserted control rod the deviation may partly be explained by the more sophisticated treatment of the control rod effect on local cross sections in TRAB-3D.

Validation calculations for TRAB-3D were continued with the NEACRP 3-D LWR core transient benchmark (Finnemann & Galati 1991) which have earlier been analyzed with the one-dimensional code TRAB. The problems consist of simulation of the core behavior after the rapid ejection of a control assembly from an initially critical core. TRAB-3D results are compared with results of the PANTHER code which are obtained by nodal calculations with 2x2 neutronic and thermal hydraulic meshes per assembly and serve as the reference solution of the benchmark (Finnemann et al. 1993).

The reactor specification of the PWR problem includes radial and axial reflectors around the active core and as in the two-dimensional calculations the reflectors were replaced by albedo boundary conditions. Axial boundary conditions were determined from the relation between albedos and two-group diffusion parameters for an infinite reflector in plane geometry. Radial albedos were adjusted to give a sufficient agreement with the reference solution of PANTHER in the initial critical states. With these modifications of the problem the deviations between TRAB-3D and PANTHER were generally less than 2 % in assembly powers in the initial state. The agreement in critical boron concentrations was comparable to the results of other codes (Finnemann et al. 1993).

The benchmark problem includes six cases of rod ejection transients starting from hot zero power or full power state of the core with different initial configurations of control rods. The ejected rod is a central rod (cases A1,2) or a peripheral rod (cases C1,2) or a group of four peripheral rods (cases B1,2) are ejected simultaneously. One and four neutronic and hydraulic meshes per fuel assembly were used in calculations of the first two cases in which a central rod is ejected. Other calculations were made only with one mesh per fuel

1	2	3										
442	381	288										
- 3	- 5	-1 3										
4	5	6	7									
891	651	533	430									
1 2	-1 2	-1 7	2									
8	9	10	11	12	13	14						
972	1 115	982	664	581	450	326						
3	3	0	-2 4	-1 5	-1 1	-2 3						
15	16	17	18	19	20	21	22					
1 132	1 227	1 132	1 115	1 007	720	568	443					
3	- 6	- 5	6	- 4	-1 4	-1 7	-2 3					
23	24	25	26	27	28	29	30	31				
832	1 200	1 314	1 143	1 208	1 105	987	831	502				
2 2	- 1	9	0	8	- 8	- 6	-2 5	-2 8				
32	33	34	35	36	37	38	39	40	41			
824	1 115	1 320	1 357	1 100	1 218	1 012	1 051	856	462			
2 7	- 9	1 5	1 2	- 4	1 6	- 7	- 7	-1 9	-1 8			
42	43	44	45	46	47	48	49	50	51	52		
1 215	1 321	1 139	1 333	1 353	1 084	1 250	1 196	1 032	604	341		
1	- 7	- 6	1 4	1 3	-1 1	1 4	- 6	- 7	-2 4	-2 5		
53	54	55	56	57	58	59	60	61	62	63		
1 053	1 307	1 329	1 126	1 252	1 340	1 178	1 264	1 176	737	475		
- 8	- 6	- 3	-1 1	- 2	0	0	7	1	- 8	- 6		
64	65	66	67	68	69	70	71	72	73	74		
921	1 238	1 217	852	869	1 263	1 212	1 338	1 171	1 063	612		
2 1	-1 0	- 1	2 1	1 9	- 1	4	1 3	- 2	3	-1 2		
75	76	77	78	79	80	81	82	83	84	85	86	
923	1 232	1 045	851	851	1 269	1 337	1 179	1 286	1 145	718	461	
1 9	4	-1 2	2 3	2 1	-1 0	1 1	9	8	1	-1 4	3	
87	88	89	90	91	92	93	94	95	96	97	98	99
1 234	1 197	1 310	1 197	1 119	1 321	1 366	1 126	1 263	1 218	1 018	577	319
- 2	-1 6	-1 0	- 2	-1 2	- 4	- 4	3	8	- 5	1	-2 2	- 3
100	101	102	103	104	105	106	107	108	109	110	111	112
1 212	1 291	1 219	1 076	1 224	1 138	1 312	1 074	1 214	1 211	1 106	786	409
- 6	0	-1 2	- 5	8	-1 4	- 2	- 3	- 5	- 5	- 3	- 3	- 8
113	114	115	116	117	118	119	120	121	122	123	124	125
835	1 067	1 284	933	929	1 272	1 236	837	822	1 160	1 111	878	451
2 4	-1 0	- 1	1 9	1 0	- 1	-1 5	2 5	1 7	- 7	1 1	1	1

Figure 18. Relative assembly powers of TRAB-3D and relative deviations (%) from POLCA4 in 90 degree sector of an Olkiluoto reactor.

assembly. Fig. 19 shows the radial fission power distribution in the asymmetric case C1 at the time of maximum power.

Fig. 20 and 21 give a comparison of time behavior of total reactor power calculated with TRAB-3D to the PANTHER results during the rod ejection transients of cases A1 (hot zero power) and A2 (full power). The effect of nodal division on the results is similar by both codes, but TRAB-3D tends to give somewhat higher values for the power maxima in the full power case. According to TRAB-3D calculations the number of hydraulic meshes per assembly has a negligible effect on the results which indicate that the observed effect of mesh number is due to the solution of the neutronic equations.

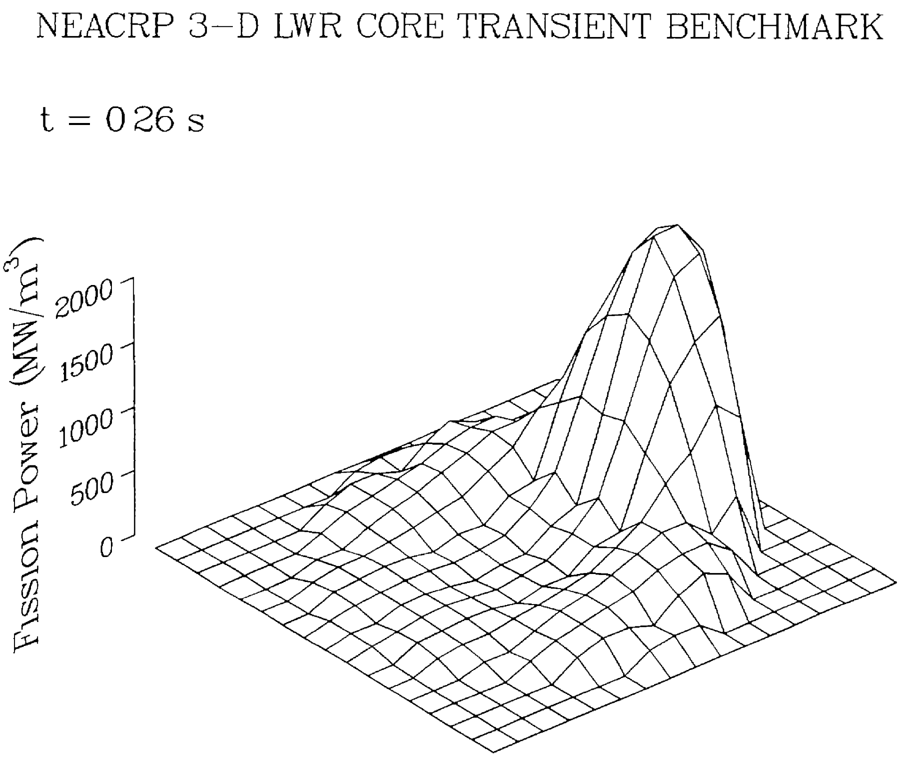


Figure 19. TRAB-3D results for the NEACRP 3-D LWR core transient benchmark C1, fission power at the time of the power peak.

Agreement between TRAB-3D and PANTHER results for the other cases B and C is roughly the same as for the case A. Especially for the zero power transients, the curves of total power by the codes agree better than the results of most other codes used in the benchmark calculations. In the full power cases A2 and C2 TRAB-3D calculates slightly higher powers than the majority of the codes, though the results are within the range of all calculations. PANTHER calculations with 1 or 4 meshes per assembly do not differ much from each other in cases B and C and also the TRAB-3D results with 1 mesh are close to the PANTHER results with 4 meshes.

Curves of maximum fuel centerline temperature in the core are shown in Fig. 22 and 23 for transients A1 and A2. TRAB-3D predicts monotonously higher values for the maximum and also for the core averaged fuel temperature than PANTHER with the same number of meshes. Since this trend is seen already in the initial state of the full power case it cannot be explained by different amount of energy released during the transient. At full power the temperature gradient in a fuel pellet is very large. Thus differences in solution method of the heat conduction equations and in calculation of average and maximum temperatures can easily lead to deviations of tens of degrees.

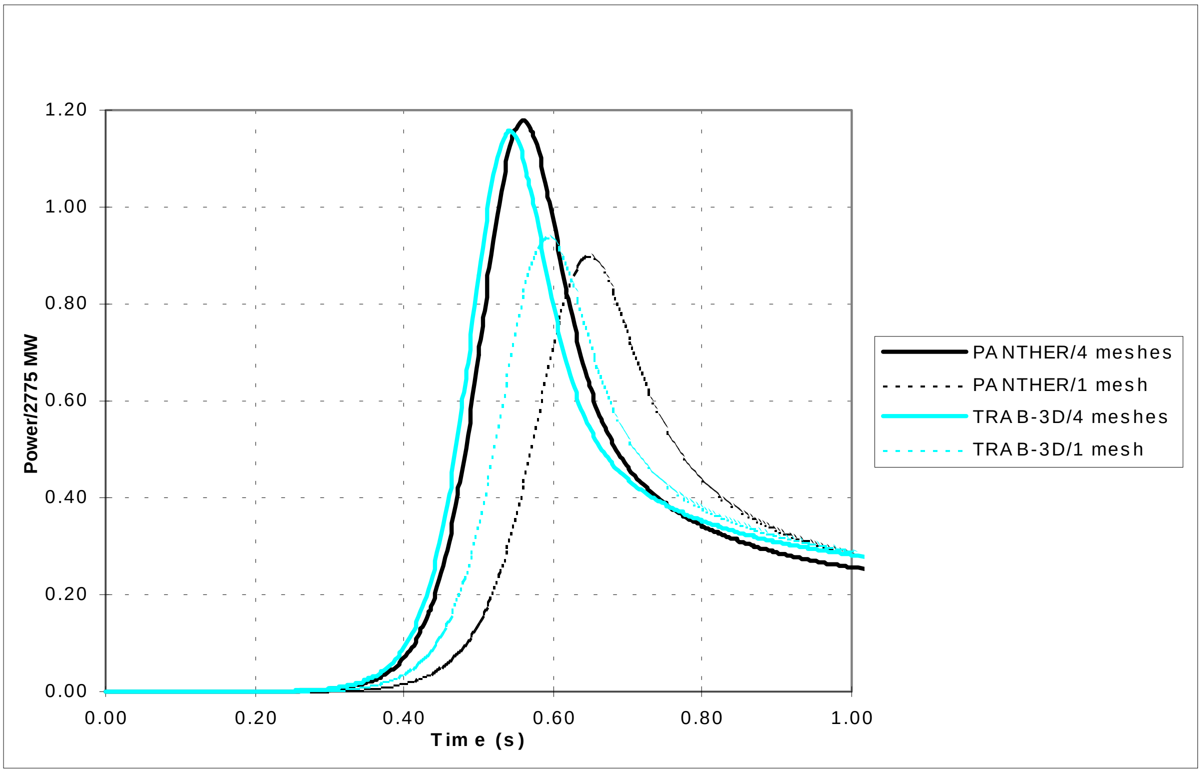


Figure 20. PWR rod ejection transient Case A1, total reactor power.

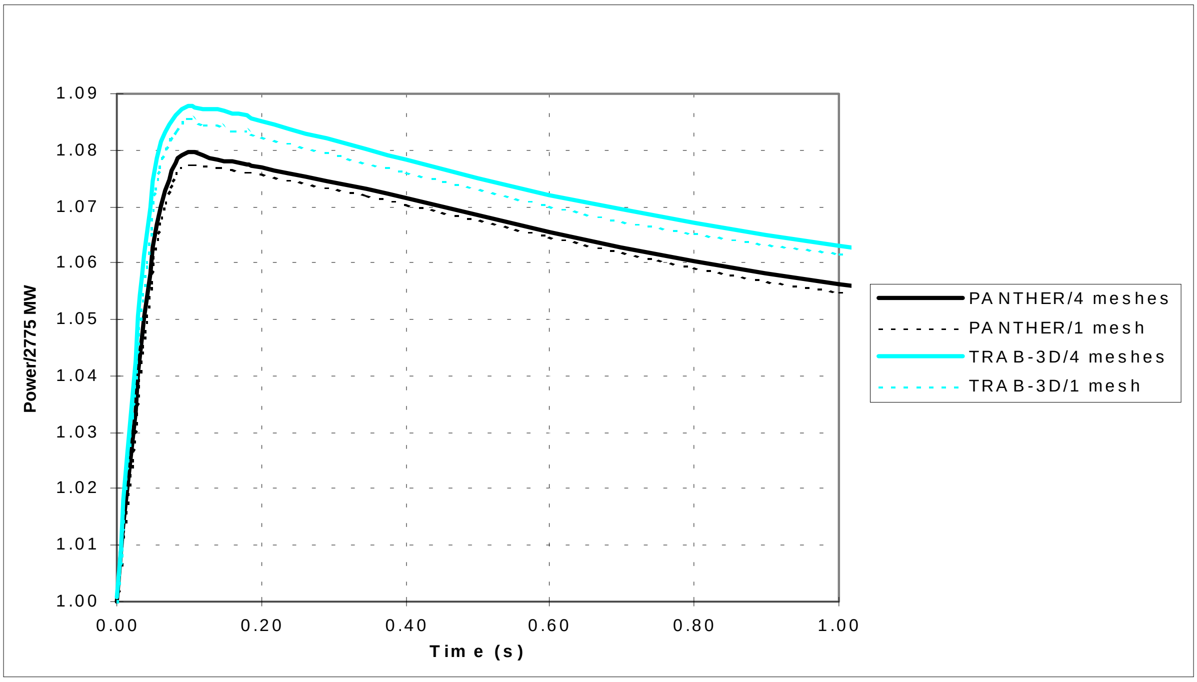


Figure 21. PWR rod ejection transient Case A2, total reactor power.

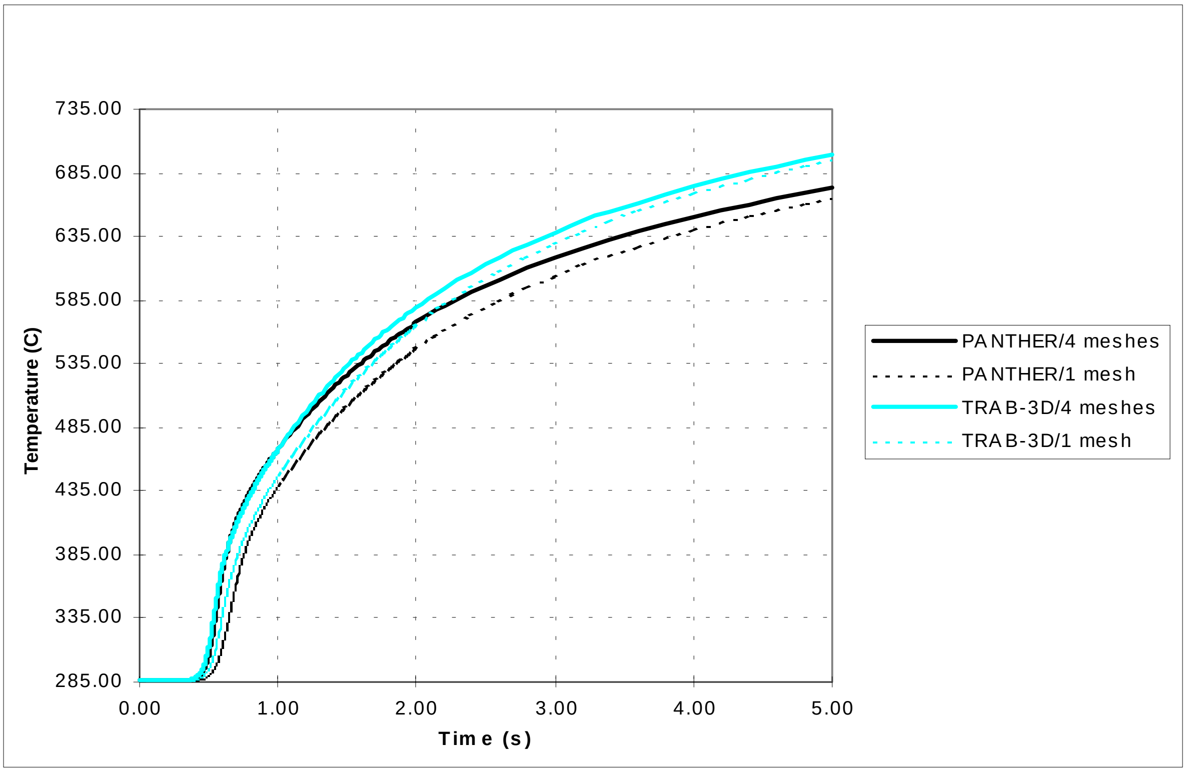


Figure 22. PWR rod ejection transient Case A1, maximum fuel center line temperature.

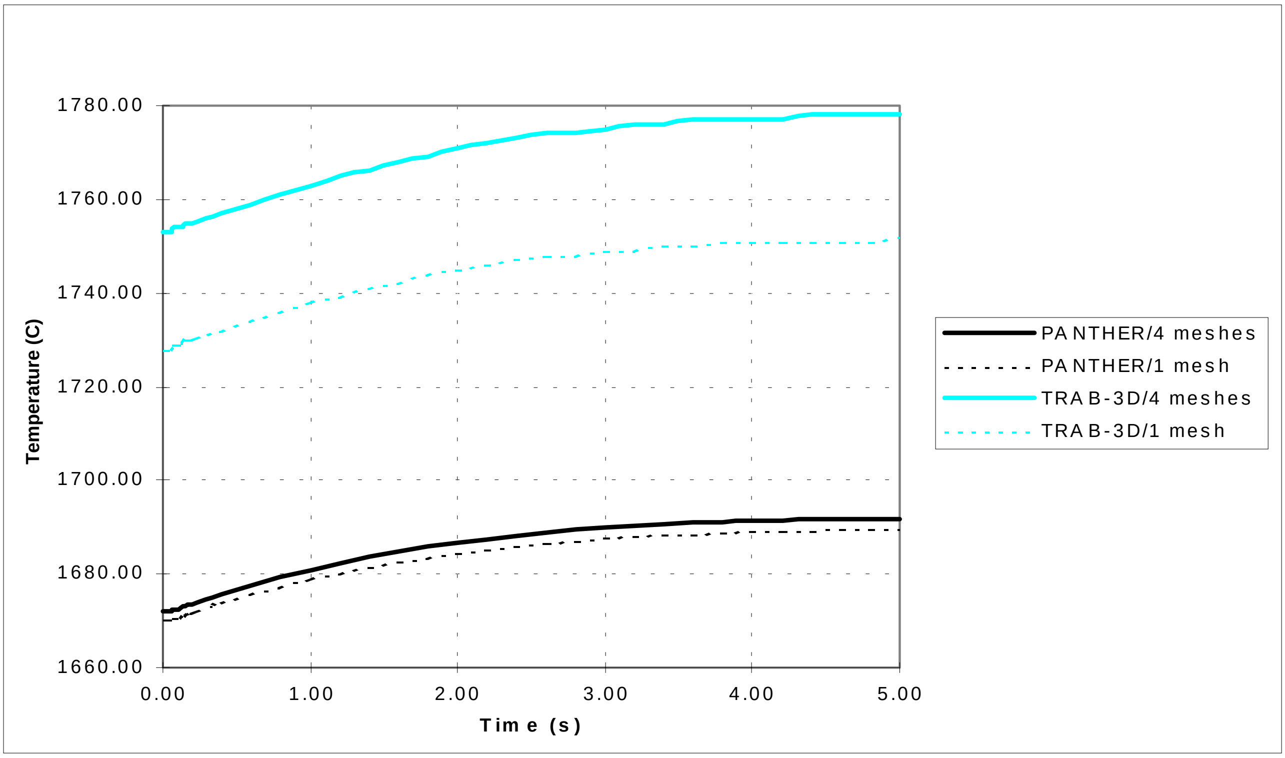


Figure 23. PWR rod ejection transient Case A2, maximum fuel center line temperature.

Another set of benchmark calculations was performed with the NEA-NSC 3-D PWR core transient benchmark of uncontrolled withdrawal of control rods at zero power (Fraikin 1993). The reactor specification is the same as in the previous PWR benchmark and the problem consists of four cases where one or two banks of control rods are withdrawn. All cases were calculated, and the preliminary results agree reasonably well with other published results (Fraikin 1996).

The overall performance of the TRAB-3D code in three-dimensional dynamic analyses for reactor cores has been successfully confirmed in calculations for NEACRP benchmark problems. The core model will next be coupled to VTT's existing BWR (and PWR) circuit models, enabling realistic testing and analyses of dynamic behavior of BWR reactors.

The validation of TRAB-3D will continue with comparisons against plant data of startup and transient measurements of the Finnish TVO reactors. Further NEA benchmark calculations include the BWR benchmarks (an inlet subcooling transient and a core pressurization transient) and participation in the planned benchmark on a PWR loss-of-flow transient.

3.2.7 Three-dimensional hexagonal dynamics benchmarks

A series of four three-dimensional hexagonal dynamic benchmark problems has been defined during 1992 - 1996 in AER. The first three problems are asymmetric control rod ejections from low power level in a realistic VVER-440 core with fuel of three different enrichments and given nuclear data. However, the complexity of the models needed in the benchmark solutions has been continuously increased. There have been two reasons for this strategy: firstly, to be able to gradually clarify the reasons of the possible discrepancies of the results, and secondly, to maximize the number of possible participants because in the beginning all the codes did not have all models available (e.g. no thermal hydraulics included). The fourth benchmark is a reactivity accident initiated by a local boron dilution, calculated with own nuclear data of each participant.

The definitions and numerous articles on the benchmark results have been published in the AER symposium series (Vidovszky (ed.) 1996). In addition, the summary of all the control rod ejection results was published in the PHYSOR 96 conference (Kyrki-Rajamäki et al. 1996).

The following codes have been used in the calculation of the benchmarks:

- DYN3D of Research Centre Rossendorf, Germany,
- HEXTRAN of VTT Energy, Finland,
- KIKO3D of KFKI Atomic Energy Research Institute, Hungary,

- BIPR8 of Russian Research Center "Kurchatov Institute", Russia, (as a separate kinetic code without thermal hydraulic feedback modelling or as coupled with the ATHLET code of GRS, Germany),
- APROS of VTT Energy, Finland.

In addition to the solutions calculated in the above mentioned organizations, independent solutions have been calculated with the code DYN3D also in KAB Berlin, Germany, and in Nuclear Research Institute Rez, Czech Republic.

The first problem was a pure reactor kinetics problem with no feedback effects included.

The second problem was similar to the first one, but the ejected control rod worth is 2 $ which results in a sharp and high power peak connected with large deformations of flux shape giving a serious test for both codes and methods of calculation. The Doppler effect is the main feedback effect for this type of transients and it is included as the only feedback.

The third problem was defined by VTT in 1994. It includes real thermal hydraulics modelling of the core. The transient itself is similar to the transient in the second problem. The reactor trip is not included in the calculation so that a new steady state is achieved after the transient.The ejected control rod reactivity worth is given because the effects of its differing values have already been studied in the first and second problems - the third problem is mainly meant for studying the thermal hydraulics effects. The core coolant flow level is assumed to be only half of the nominal level so that boiling is occurring during the transient. A hot channel calculation with departure from nucleate boiling and film boiling calculations is also included in the problem.

The third problem was calculated in five countries with four different codes (HEXTRAN, BIPR8/ATHLET, DYN3D and KIKO3D). The prompt fission power peak was predicted as almost the same by all codes due to the fixed key parameter of ejected rod reactivity worth, but after that there were immediately deviations in the maximum fuel and cladding temperatures due to the different heat transfer rate to the coolant (Fig. 24 and 25).

During the power burst after the control rod ejection there are large deviations in the time behaviour of the fuel and coolant temperatures and densities near the ejected control rod. On the other hand the power distributions are fairly close to each other especially at the early phase of the transient when the coolant feedback is not yet significant and at the later phase when the reactor approaches a new steady state. However, rather large local deviations appear in the power distributions of some assemblies at the time of maximum boiling of coolant. Radial distributions calculated by HEXTRAN for fission power at time of the power peak and for void fraction at time of maximum outlet void fraction are seen in Fig. 26 and 27.

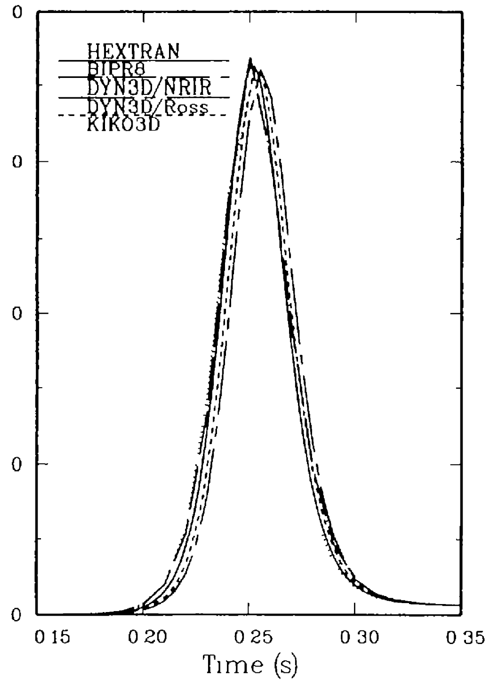


Figure 24. Total fission power during power peak.

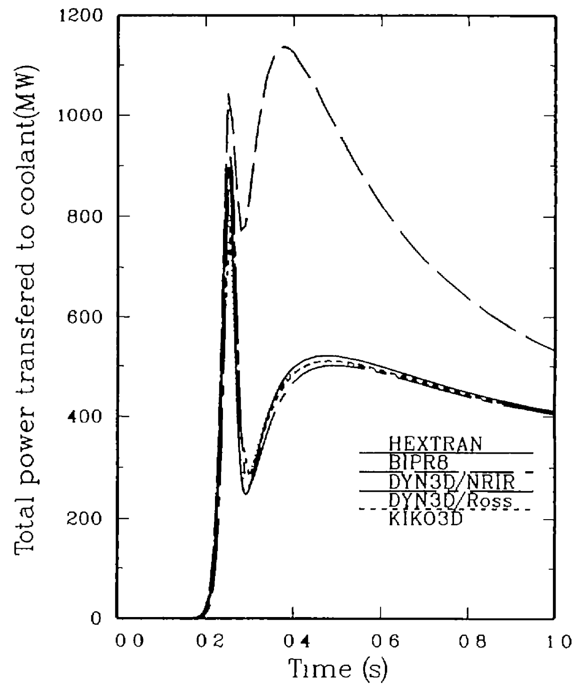


Figure 25. Total power transferred to coolant during first second.

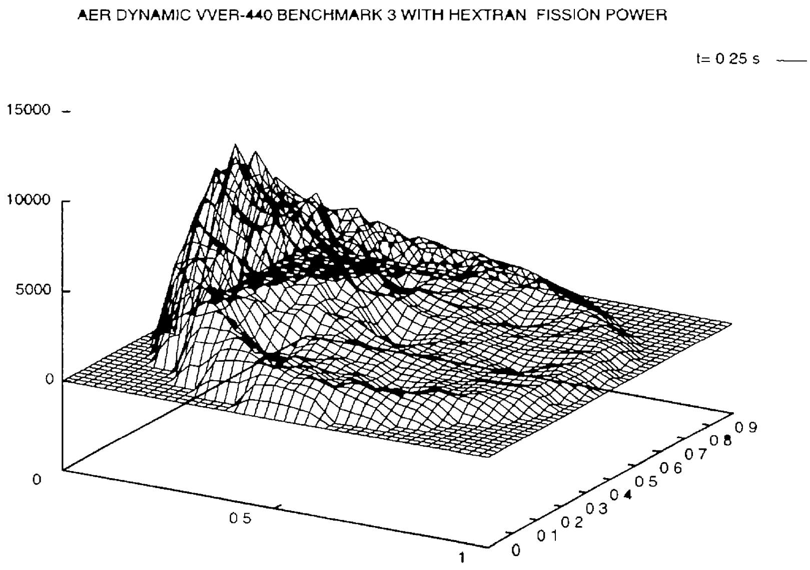


Figure 26. HEXTRAN results for the third dynamic AER benchmark, fission power (MW/m^3) at time of the power peak.

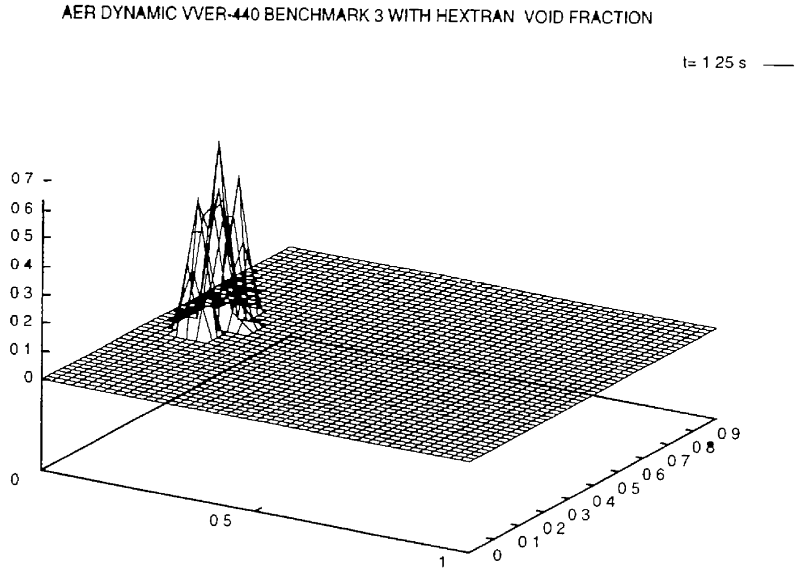


Figure 27. HEXTRAN results for the third dynamic AER benchmark, void fraction at time of maximum outlet void fraction.

All thermal hydraulics models suggested in the benchmark definition were not used by the participants. For instance, using of different temperature dependent gas gap conductivity data changes the results dramatically. Also the film boiling and its collapse were calculated with different correlations, which caused large differences in the hot channel temperatures above critical heat flux.

The benchmarks have been useful. The reliability of the neutron kinetics parts of the nodal codes using different methods has been confirmed. Also the thermal hydraulics results of HEXTRAN and DYN3D are very similar. The work gives adequate basis to continue benchmarking mostly concentrating on the interaction of the thermal hydraulics and fission power distribution behaviour.

The fourth benchmark formulated by VTT (Kyrki-Rajamäki 1996) is a reactivity accident initiated by a local boron dilution. First preliminary results of the benchmark were compared in the working group meeting in April 1997. Most participants had developed a new special boron tracking method to get reasonable results. A comparison paper will be prepared to the AER symposium in September 1997. The fifth benchmark is aimed to be a steam line break and include also the cooling circuit models.

3.2.8 Improvements and studies in hot channel modelling

Traditionally in safety analysis hot channel calculations are made by considering the hot rod and the surrounding flow channel separated from the adjacent rods and flow channels, which means that no advantage is taken from transverse mixing and the flow in the channel is determined only by pressure difference over the core. This method is supposed to give a conservative estimate of the state of the hottest fuel rod. The separated channel method may be overly conservative in nominal flow conditions, while the conservativity is not quite so clear in natural circulation.

In order to evaluate the conservativity in various conditions, subchannel and separated channel calculations were conducted parallelly with the COBRA-IV-I program. The separated hot channel model proved to be conservative in all conditions (Piponius 1995). The separated channel calculation of COBRA was also verified by comparing it to the results obtained from the TRAB reactor dynamics program. The comparisons showed only small differences between the programs, which indicates the ability of COBRA for hot channel calculations in safety analyses (Piponius 1996a). The code has already earlier been modified by VTT, and during this work new properties, such as time dependent axial power profile and oxidation model, have been added to the code (Piponius 1996b).

The main objective of the future work is to increase the accuracy of the subchannel calculations, while at the same time ensure that the results stay conservative. In order to improve numerical performance, application of the Piecewise Linear Interpolation

Method (PLIM) is currently being investigated by first studying the axial flow equations of COBRA-IV-I and by searching for a proper presentation for transverse interactions between subchannels. At the same time modifications to the current COBRA program are continued. More accurate data on crossflow mixing is searched and possibly comparisons are made against single phase results of fully 3-D flow modelling programs.

3.2.9 Application of Monte Carlo technique

Experience on application of the Monte Carlo technique has mostly been obtained in tasks outside the project, and the work carried out in the project has been closely connected to these tasks. The MCNP Monte Carlo code has been applied to gamma and neutron dose calculations, criticality safety studies and for fusion neutronics calculations.

Version 4A of MCNP has been installed along with data libraries based on ENDF/B-V and VI. Calculations of neutron and gamma fluxes and heat rates at various locations in and near the equatorial and divertor ports were carried out for the ITER project using the MCNP4A and the FENDL-1 cross section library. Special attention was paid to the neutron flux behind an ICRH heating module, which is important to the design of vacuum windows for the transmission lines.

Assistance was rendered to VTT Chemical Technology in their calculations on the use of a TRIGA reactor for Boron Neutron Capture Therapy (BNCT). A detailed MCNP model for a TRIGA core was set up. A version of MCNP4A with the correlated sampling option (kcorr) for criticality calculations was acquired and implemented. The calculation system was tested by calculating the TRIGA control rod worths. The agreement between the calculated and measured rod worths was good.

The core model was used to design such fuel loading patterns where the flux towards the BNCT-column is optimized. In a loading pattern designed by VTT Energy 3,9 % of the neutrons is reaching the BNCT-column, which is an essential increase compared with the present loading pattern where the corresponding value is 3,0 %. This should mean 25 % shorter treatment times.

3.2.10 Improved formalism for the two-group cross section sets

The data which describes the neutron physics processes in the reactor core are calculated with fuel assembly burnup programs. The data sets consist of two group cross sections and other parameters of interest. A linking program is needed to process these data into a formalism accepted by a core simulator.

At present, the CRFIT postprocessor program is used as a link between the CASMO-HEX fuel assembly burnup code and the core simulators. The reactivity feedback models in

CRFIT are based on the polynomials of first or second order. The coefficients of the polynomials are determined simply by demanding that a polynomial produces the CASMO-HEX results at the calculated state points, the number of which is therefore very limited. The CRFIT model is, however, adequate for static reactor physics studies, where the variation of the state variables is rather small. This assumption is not valid in many time-dependent safety studies. Then a higher order formalism is needed.

Development of a new two-group data formalism was started during 1995 in connection with TRAB-3D development. The final goal is to modernize all subroutines of the HEXBU, HEXTRAN and TRAB-3D codes dealing with the reactor physics. The new postprocessor code is based on a nonlinear least square method. It will provide full generality with respect to:

- choice of the feedback function forms (e.g the order of polynomials),
- number of feedback variables,
- number of the state points,
- order of the state points.

The least square method has been tested in connection with a contract work concerning the calculation of the two-group constants of a VVER-440 bundle for safety studies, where the variation of the relevant feedback variables was large. The results of the test were promising. Some preliminary studies indicate that also the effects of the history variables could be presented as fitting functions.

3.2.11 Generation of program-wise nuclear data libraries

The NJOY nuclear data processing system is a comprehensive computer code package for producing continuous-energy and multigroup neutron data from the evaluated nuclear data libraries. The NJOY-94.35 and TRANSX-2.15 nuclear data processing codes were installed, and applied to generate MCNP and ANISN neutron data from the ENDF/B-VI.3 and JEF-2.2 evaluated nuclear data libraries. The processing codes and the nuclear data were acquired from NEA Data Bank.

NJOY-94.35 was applied to generate continuous-energy neutron data at 300, 600 and 900 K for the MCNP Monte Carlo transport code from the evaluated nuclear data libraries ENDF/B-VI.3 and JEF-2.2. The reliability of the generated MCNP neutron data was tested by a number of MCNP4A simulations. The ENDF/B-VI.3 -based 300 K neutron data were compared to a standard MCNP neutron data library ENDF60. The comparison was based on the calculation of the infinite medium multiplication factor for three different fuel bundles. It was found that the calculated infinite medium multiplication factors using the ENDF/B-VI.3 -based 300 K neutron data were 150 - 350 pcm higher than those obtained using ENDF60. The new evaluation for U-235 in the third revision of ENDF/B-VI accounts for the difference.

The Doppler coefficient of reactivity is a crucial parameter in safety analyses for the transients in light water reactors. The high temperature MCNP neutron data were applied to perform a set of benchmark calculations for the Doppler coefficient of reactivity for UO2 and MOX fuel (Mosteller & Eisenhart 1991, Holly et al. 1991). R. D. Mosteller et al. performed the benchmark calculations using MCNP3A code and ENDF/B-V -based neutron data. At VTT the calculations were performed with MCNP4A. The ratio of the calculated infinite multiplication factors using ENDF/B-VI.3 -based data to the reference results varied from 1.004 to 1.006. The ratio of the calculated infinite multiplication factors using JEF-2.2 -based data to the reference results varied from 1.006 to 1.008. The calculated Doppler coefficients of reactivity were in a relatively good agreement with the reference results (Mosteller & Eisenhart 1991, Holly et al. 1991).

NJOY-94.35 was applied to generate groupwise neutron data in MATXS format from the ENDF/B-VI.3 evaluated nuclear data library. The group structure and the weight spectrum were those of the VITAMIN-B6 (White et al. 1996) fine-group library. TRANSX-2.15 was applied to generate transport tables for the ANISN one-dimensional multigroup transport code. The above mentioned benchmark calculations for the Doppler coefficient of reactivity for UO2 fuel were performed using ANISN and the new ENDF/B-VI.3 -based transport tables. The ANISN infinite medium multiplication factors were about 1000 pcm higher than the MCNP4A results. This is acceptable considering the differences in the calculation methods. The calculated Doppler coefficients of reactivity for UO2 fuel are shown as a function of the fuel enrichment in Fig. 28. The calculated ANISN Doppler coefficients of reactivity were in a good agreement with the MCNP results.

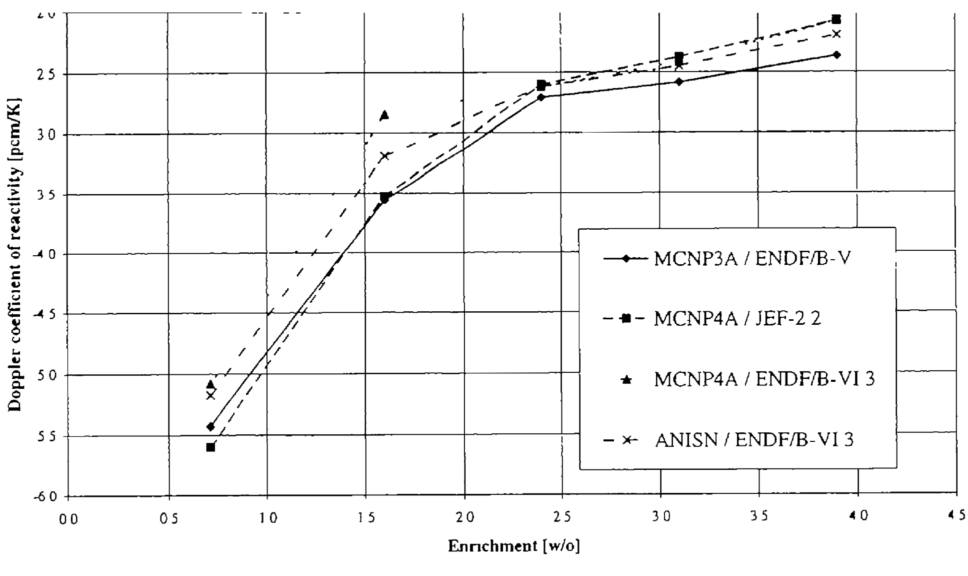


Figure 28. Doppler coefficient using different codes and nuclear data libraries.

3.2.12 Core loading optimization

CORFU is a BWR core reload design program developed at VTT (Höglund 1992, Höglund et al. 1995). With the successful implementation of a method for optimization of control rod patterns (Heino 1996, Heino 1997), all four main modules of the code are now operational, although many improvements would still be desirable, e.g. concerning user-friendliness. CORFU has been tested on the Olkiluoto 1 reactor (Höglund & Solala, 1997) and similar trials on the Swedish Oskarshamn 3 reactor are under way.

For the 1995 loading of Olkiluoto 1, the CORFU1 module produced a preliminary loading pattern that was already 66 % correct, i.e. with the right bundles in the right positions, (Fig. 29). The control rod pattern optimization of the CORFU2 module is based on a simple model for making fast estimates of the effect control rod movements on k_{eff} as well as on power, heat flux or linear heat rating and dryout margin distributions. As an example, Fig. 30 shows the original vertical power distribution in the core (computed by POLCA) corresponding to a certain control rod pattern together with the CORFU2 prognosis and the new POLCA result after the pattern has been changed. The two latter curves are in very good agreement and even in a control cell the predicted change in the vertical power shape of an individual fuel bundle is usually accurate enough for fast comparison of many different control rod patterns as shown in the right part of the figure. The CORFU3 module, which suggests bundle moves to improve the loading scheme, must usually be applied at least 10 - 20 times in order to achieve acceptable results. Some purely manual operations are still often useful between the CORFU3 runs.

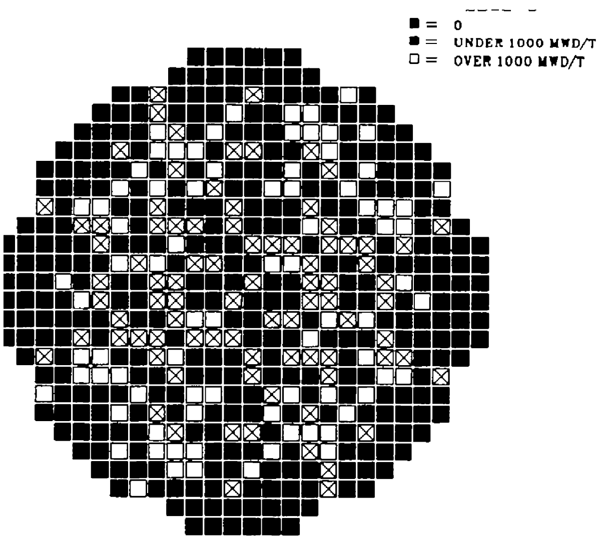
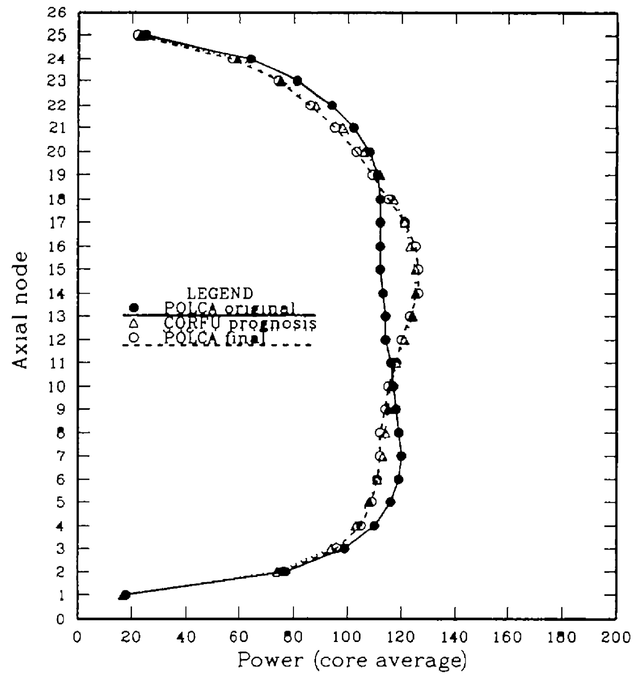


Figure 29. Difference in horizontal BOC burnup distribution between the final loading and CORFU's first preliminary loading pattern. Black square: right bundle in right place.



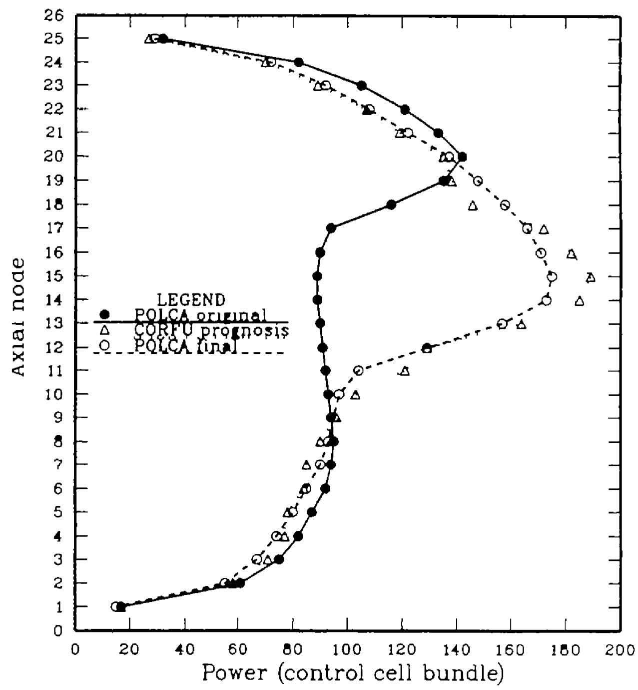


Figure 30. Vertical power distribution before and after a change in the control rod pattern together with the "after" distribution predicted by the CORFU2 module. The whole-core average is shown in the upper figure and the results for an individual fuel bundle in a control cell in the lower figure.

3.2.13 Updating and documentation of reactor physics programs

The reactor physics code system has been upgraded and its documentation has been improved to increase the user-friendliness of the codes.

Based on an agreement between the Studsvik Core Analysis AB (SCOAB) and VTT Energy the trial use of the advanced three-dimensional two-group reactor analysis code SIMULATE-3 of SCOAB was started at VTT in 1993. The careful simulations of the whole operating history of both TVO reactors were carried out in 1993 - 1995. In 1995 TVO made a license agreement with Studsvik concerning the SIMULATE-3 program package. Based on the Studsvik/TVO agreement, VTT Energy made an agreement with TVO to obtain a SIMULATE sublicence with the same rights and obligations as TVO. The SIMULATE program package was delivered in the beginning of 1996.

Validation of the CASMO/SIMULATE code package for the TVO reactors has been reported in several conferences (Höglund et al. 1995, Roine et al. 1995). A new application at VTT has been to calculate rod power histories with CASMO-4/SIMULATE-3 for the fuel performance code ENIGMA (Kelppe et al. 1995).

IVO Power Engineering Ltd and VTT Energy made an agreement on the development of a hexagonal version of the CASMO-4 fuel assembly burnup code with the Studsvik Core Analysis AB in 1995. The first version of the hexagonal CASMO-4 was delivered in June 1996. In the end of 1996 VTT Energy acquired also a CASMO-4 version capable of handling cluster geometries (Knott et al. 1997).

The code CROCO for collapsing three-dimentional group constants to one dimension has been updated and documented, and used to produce input data for transient analyses.

The documentation of the nodal two-group code HEXBU-3D/MOD5 has been supplemented with an updated input manual, serving also as documentation for the neutronics part of the dynamics code HEXTRAN. A user's manual for the hexagonal fuel assembly program CASMO-HEX has been written.

The VENUS-1 Two-Dimensional Benchmark on Ex-Core Dosimetry Computations of the NEA/NSC Task Force on Computing Radiation Dose and Modelling of Nuclear Radiation-Induced Degradation of Reactor Components was calculated using the REPVICS program system, mainly DORT, version 2.8.14, and the BUGLE-93 and BUGLE-96 libraries. A series of calculations using different approximations was performed. This showed which approximations had the greatest influence on the results and thus where improvements would be most useful if we wish to prepare a new kernel library of PREVIEW, the program currently used in Finland for calculations of reactor pressure vessel fluence.

The latest version (4.3) of the SCALE program package has been installed on VTT's work stations to be used mainly as a tool for criticality safety studies.

References for Section 3.2

Finnemann, H., Bauer, H., Galati, A. & Martinelli, R. 1993. Results of LWR core transient benchmarks. Paris 1993. OECD Nuclear Energy Agency (NEA), Nuclear Science Committee (NSC). NEA/NSC/DOC(93)25. 81 p.

Finnemann. H. & Galati, A. 1991. NEACRP 3-D LWR core transient benchmark. Paris 1991. OECD Nuclear Energy Agency (NEA), Committee on Reactor Physics (CRP). NEACRP-L-335 (Revision 1), October 1991 (January 1992).

Fraikin, R. 1993. NEA-NCS 3-D PWR core transient benchmark. Uncontrolled withdrawal of control rods at zero power. Paris 1993. OECD Nuclear Energy Agency (NEA), Nuclear Science Committee (NSC). NEA/NCS/DOC(93)25-9, September 1993.

Fraikin, R. 1996. Review of a NEA-NCS PWR benchmark on uncontrolled withdrawal of control rods at zero power. Proceedings of the PHYSOR '96, International Conference on the Physics of Reactors -Breakthrough of Nuclear Energy by Reactor Physics-. September 16 - 20, 1996, Mito-city, Ibaraki-ken Japan. Mito: Japan Atomic Energy Research Institute. Vol. 3, pp. J-99 - J-108.

Heino, J. 1996. Development of a computer program for optimization of control rod patterns in a boiling water reactor (in Finnish). Technical Report RFD-36/96. 23 p. + app. 23 p. (Kiehutusvesireaktorin säätösauvakuvioiden optimointiohjelman kehittäminen.)

Heino, J. 1997. On the accuracy of CORFU2 and changes made in the code (in Finnish). Technical report RFD-3/97. 5 p. + app. 8 p. (CORFU2:n tarkkuudesta ja siihen tehdyistä muutoksista muutoksista.)

Holly, J.T., Mott, L.A & Mosteller, R.D. 1991. Benchmark Calculations for the Doppler Coefficient of Reactivity in Mixed-Oxide Fuel. American Nuclear Society, 1991. Proceedings of the International Topical Meeting on Advances in Mathematics, Computations, and Reactors, April 28 - May 2, 1991 Green Tree Marriot, Pittsburgh, PA, USA, Vol. 2, pp. (9.2 1-1) – (9.2 1-12).

Höglund, R. 1992. The CORFU program for BWR core reload design. In: In-core Fuel Management: Reloading Techniques. Proceedings of a Technical Committee Meeting and Workshop, Vienna 19 - 21 October 1992. Vienna: International Atomic Energy Agency. Pp. 85 - 92. (IAEA-TECDOC-816.)

Höglund, R., Roine, T. & Solala, M. 1995. Experience in the utilization of BWR fuel including core follow calculation with the CASMO-SIMULATE code package and efficient core design with the CORFU program. Transactions of the International KTG/ENS TOPical Meeting on Nuclear Fuel TOPFUEL '95. March 12 - 15, 1995 Würzburg, Germany. Volume II, Poster Papers. Pp. 161 - 164.

Höglund, R. & Solala, M. 1997. ICFM for a BWR using the CORFU core designer. Proceedings of the Topical Meeting Advances in Nuclear Fuel Management II. March 23 - 26, 1997, Myrtle Beach, SC, USA. American Nuclear Society. Vol 2, pp. 17-13 - 17-20. (EPRI TR-107728-V1.)

Kaloinen, E. & Kyrki-Rajamäki, R. 1997. TRAB-3D, a new code for three-dimensional reactor dynamics. CD-ROM Proceedings of ICONE-5, 5th International Conference on Nuclear Engineering. "Nuclear Advances through Global Cooperation". May 26 - 30 1997, Nice, France. Paper ICONE5-2197.

Kelppe, S., Roine, T. & Lunabba, R. 1995. ENIGMA calculations on fuel irradiated in TVO I reactor compared with pool-side fission gas release measurements. Transactions of the International KTG/ENS TOPical Meeting on Nuclear Fuel TOPFUEL '95. March 12 - 15, 1995 Würzburg, Germany. Volume II, Poster Papers. Pp. 109 - 112.

Knott, D., Edenius, M., Peltonen, J. & Anttila, M. 1997. Results of modelling hexagonal and circular cluster fuel assembly designs using CASMO-4. Proceedings of the Topical Meeting Advances in Nuclear Fuel Management II. March 23 - 26, 1997, Myrtle Beach, SC, USA. American Nuclear Society. Vol 1, pp. 5-1 - 5-15. (EPRI TR-107728-V1.)

Kyrki-Rajamäki, R. 1995a. On the role of burnup effects of fuel properties in RIA analyses. Proceedings of the Specialist Meeting on Transient Behaviour of High Burnup Fuel (NEA/CSNI and CEA/IPSN), September 12 - 14, 1995, Cadarache, France. To be published.

Kyrki-Rajamäki, R. 1995b. Three-dimensional reactor dynamics code for VVER type nuclear reactors. Espoo: Technical Research Centre of Finland. 51 p. + app. 80 p. (VTT Publications 246.) (DTech thesis).

Kyrki-Rajamäki, R. 1996. Definition of the 4th dynamic AER benchmark. Proceedings of the 6th symposium of the AER on VVER Reactor Physics and Reactor Safety. September 23 - 26, 1996, Kirkkonummi, Finland. KFKI/AEKI, Budapest. Pp. 237 - 246.

Kyrki-Rajamäki, R., Grundmann, U. & Keresztúri, A. 1996. Results of three-dimensional hexagonal dynamic benchmark problems for VVER type reactors. Proceedings of the PHYSOR '96, International Conference on the Physics of Reactors. Breakthrough of Nuclear Energy by Reactor Physics. September 16 - 20, 1996, Mito-city, Ibaraki-ken Japan. Mito: Japan Atomic Energy Research Institute. Vol. 3. Pp. J-229 - J-238.

Kyrki-Rajamäki, R. & Stenius, T. 1995. Calculation of local boron dilution accidents with HEXTRAN code. Proceedings of the ANS Topical Meeting: International Conference on Mathematics and Computations, Reactor Physics, and Environmental Analyses. April 30 - May 4, 1995, Portland, Oregon, USA. La Grange Park, Ill, USA: American Nuclear Society. Vol. 1, pp. 274 - 283.

Mosteller, R.D. & Eisenhart L.D. 1991. Benchmark Calculations for the Doppler Coefficient of Reactivity. Nuclear Science and Engineering, 1991. Vol. 107, pp. 265 - 271.

Narumo, T. 1997. Modelling of two-phase flow based on separation of the flow according to velocity. DTech thesis. To be published.

Narumo, T. & Rajamäki, M. 1995. Comparing the SFAV-based two-fluid model with virtual-mass models. International Journal of Power and Energy Systems, 1995. Vol. 15, No. 1, pp. 29 - 36.

Narumo, T. & Rajamäki, M. 1996a. Interfacial drag and dividing phases in two-phase modelling. OECD Workshop on Transient Thermal Hydraulic code requirements. November 5 - 8, 1996, Annapolis, Md USA. To be published.

Narumo, T. & Rajamäki, M. 1996b. Wall and interfacial friction terms of a dynamic whole-range two-fluid model. Transactions of the ANS/ENS 1996 International Conference. Washington DC, 10 - 15 November 1996. La Grange Park, IL: American Nuclear Society. Pp. 390 - 392. (TANSAO 75 1-490).

Piponius, T 1995. Numerical simulation of cross flows between sub-channels in a rod bundle and changes caused by mixing in the conditions of a hot channel (in Finnish). Technical report RFD-20/95. 63 p. (Sauvanipun alikanavien välisten poikittaisvirtauksien numeerinen simulointi ja sekoittumisen aiheuttamat muutokset kuuman kanavan olosuhteisiin.)

Piponius, T. 1996a. Comparisons between COBRA and TRAB (in Finnish). Technical report RFD-2/96. 36 p. (COBRA:n ja TRAB:n vertailut.)

Piponius, T. 1996b. Changes made in the COBRA-IV code (in Finnish). Technical report RFD-1/96. 8 p. (COBRA-IV:ään tehdyt muutokset.)

Rajamäki, M. 1996a. Elimination of numerical diffusion in one-phase and two-phase flows. OECD Workshop on Transient Thermal Hydraulic code requirements. November 5 - 8, 1996, Annapolis, Md USA. To be published.

Rajamäki, M. 1996b. Modelling of appearance, movement and disappearance of water-level-like discontinuity in channel geometry. European two-phase flow group meeting. June 3 - 5, 1996 Grenoble, France.

Rajamäki, M. & Narumo, T. 1995. A six equation SFAV-model for two-phase flow with correct propagating velocities of disturbances. Journal of Numerical Heat Transfer; Part B, 1995. Vol. 28, No. 4, pp. 415 - 436.

Rajamäki, M. & Saarinen, M. 1994a. Accurate One-Dimensional Computation of Frontal Phenomena by PLIM. Journal of Computational Physics, 1994. Vol. 111, No. 1, March 1994, pp. 62 - 73.

Rajamäki, M. & Saarinen, M. 1994b. New two-fluid formalism: separation of the flow according to velocities. Numerical Heat Transfer, Part B, Fundamentals, 1994. Vol. 26, No. 4, pp. 439 - 454.

Rajamäki, M. & Saarinen, M. 1991. PLIM shape preserving characteristics method for flow equations. Proceedings of the International Topical Meeting on Advances in Mathematics, Computations and Reactor Physics, April 28 - May 2, 1991, American Nuclear Society. P. 12.1.1-1 - 12.1.1-13.

Roine, T., Anttila, M., Höglund, R. & Solala, M. 1995. Validation of CASMO/SIMULATE code package for TVO boiling water reactors. Proceedings of the ANS Topical Meeting: International Conference on Mathematics and Computations, Reactor Physics, and Environmental Analyses. April 30 - May 4, 1995, Portland, Oregon, USA. La Grange Park, Ill, USA: American Nuclear Society. Vol. 2, pp. 1058 - 1067.

Saarinen, M. 1994. Characteristics-based modelling of flow problems. Espoo: Technical Research Centre of Finland, Publications 164. 42 p. + app. 58 p.

Vidovszky, I. (ed.). 1996. Proceedings of the 6th symposium of the AER on VVER Reactor Physics and Reactor Safety. September 23 - 26, 1996, Kirkkonummi, Finland. KFKI/AEKI, Budapest. 696 p.

Vitanza, C. 1991. Fuel performance at high burn-up. Proc. of the ENS Topical Meeting on Nuclear Fuel TOPFUEL '95. Würzburg, March 12 - 15, 1995. Transactions, 1995. Volume I, Invited Papers, pp. 153 - 161.

White, J.E., Ingersoll, D.T., Slater, C.O & Roussin, R.W. 1996. BUGLE-96: A revised multigroup cross section library for LWR applications based on ENDF/B-VI Release 3, American Nuclear Society Radiation Protection & Shielding Topical Meeting, April 21 - 25, 1996, Falmouth, MA, USA, 1996.

3.3 Thermal hydraulics experiments and analyses for VVER and ALWR plants

Two projects cover the publicly funded experimental research in nuclear reactor thermal hydraulics. TEKOJA project concentrates on the thermal hydraulics of VVER type pressurised water reactors, such as in Loviisa, Finland. TEKOJA experiments produce data for validation of thermal hydraulic computer codes, support development of accident management measures and provide information about safety system performance in accidental situations. The second project (PAHKO) investigates passive safety injection systems of new generation nuclear reactors. The safety system examined consists of a passive safety injection tank called core make-up tank (CMT) and two pipelines. The pressure balancing line (PBL) connects the top of the CMT to one cold leg. The injection line (IL) connects the bottom of the CMT to the downcomer. PAHKO experiments provide data for evaluation of passive safety injection systems performance during accidents, and for validation of thermal hydraulic computer codes. This data is of great value since only a limited amount of public data is available, because of commercial interests of companies designing ALWR's.

PACTEL is the main experimental facility in the TEKOJA project. PACTEL is the largest and most versatile simulator of VVER type pressurised water reactors in the world. The experiments with PACTEL started during the YKÄ research program within the PROPA project. PROPA experiments included OECD/NEA/CSNI International Standard Problem ISP 33 experiments, which was the first standard problem with VVER type reactor simulator. Organisations from both OECD and non-OECD countries took part widely in ISP 33. PROPA experiments also provided valuable information about natural circulation characteristics of VVER type reactors. The PACTEL experiments showed the influences of special features of VVER reactors, such as hot leg loop seals and horizontal steam generators, on natural circulation. TEKOJA experiments have continued investigations of natural circulation characteristics of VVER reactors. Experimenters have also studied mechanisms leading to formation of non-borated water plugs to the primary circuit (problematic of boron dilution).

PAHKO experiments also continue the work started within the PROPA project. These experiments also use PACTEL. PACTEL operators changed the safety injection accumulators of the loop for the PAHKO experiments. The experiments carried out within the PROPA project showed that rapid condensation in the CMT may stop safety injection from the tank. For the PAHKO experiments PACTEL operators modified further the passive safety injection system. In the earlier experiments the safety injection system included two PBLs: one to the pressurizer, the other to one cold leg. In the new experiments PACTEL operators removed the pressurizer PBL. The reason for removal of the line was that one possible reason for rapid condensation in the CMT is the flow of water from pressurizer (or cold leg), which broke the saturated liquid layer in the CMT. The saturated water layer separates cold water from steam. Break up of the layer led to direct contact condensation in the CMT. Two first PAHKO experiment series of five experiments each are ready. The third series is planned to be run in autumn 1997.

APROS and RELAP5 computer code analyses support the experimental work. A basic PACTEL input deck for both codes is available. The project group uses the codes mainly for pretest simulation of the experiments. For example, the analysts made parametric studies of the SBLOCA experiments with the APROS 3.0 code version. The analysts evaluated the correct break sizes for the experiments. The break size is an important factor in boron dilution transients. In this task, the use of the APROS code was essential. In these pretest simulations, the analysts used the five equation model of APROS. The six equation model of APROS was in use in the pre- and posttest simulations of the PACTEL SBLOCA experiments, post-test analyses of the passive safety injection experiment (GDE-24), and pretest simulations of the second PAHKO series. A diploma thesis presents the results of the SBLOCA analyses (Plit, 1996). A code validation report includes the results of the post-test analyses for the GDE-24 experiment (Vihavainen, 1996). The next chapter includes an overview of these analyses. A test specification report summarises the results of pretest analyses of the second PAHKO series (Tuunanen and Vihavainen 1996).

The RELAP5 code was in use in the post-test simulations of PRISE experiments (Riikonen, 1996a), pre-test analyses of low pressure natural circulation, and passive safety injection experiments (Tuunanen et al. 1996). The project group also used the code in post-test simulations of the earlier passive safety injection experiment, GDE-11, and post-test analyses of Hungarian standard problem SPE-4 experiments. Later this report describes the results of simulations of the PRISE experiments and the low pressure natural circulation experiments. The results helped in selecting the experimental parameters (Semken & Tuunanen, 1996). CAMP reports give the results of the analyses of GDE-11 and SPE-4 experiments (Tuunanen 1995 and Bánáti 1995).

3.3.1 Natural circulation experiments in low primary pressure (SIR-20, SIR-21 and SIR-23)

Three experiments, SIR-23, SIR-20, and SIR-21, performed in PACTEL examined natural circulation with decreasing primary-coolant inventory. An experimental data report presents the results (Semken & Tuunanen 1996). The PACTEL single loop experiments simulated matching conditions for two isolated primary coolant loops of a Loviisa VVER-440 pressurised water reactor. The primary difference among the experiments was the beginning primary- and secondary-side pressures. VVER-440 reactor steam generators are horizontal. Western pressurised water reactor designs use vertical steam generators. The horizontal steam generator layout coupled with the hot leg piping geometry in the VVER-440 affects natural circulation behaviour during inventory decrease. Therefore, the data provided by these three experiments are of unique value.

The main objectives of the experiments were as follows.

- They should compare PACTEL behaviours with those of previously performed small and medium scale natural circulation experiments.
- They should reveal the influence of primary-side pressure on natural circulation behaviour.
- They should provide data for corroborating relevant computer models and codes.
- They should test the newly installed PACTEL steam generators.

Also, the experiments would provide more data on the thermal hydraulics associated with natural circulation in the PACTEL. First, the experiments would give mass flow data to aid in developing equations for predicting mass flow rate during changing natural circulation modes. Next, the behaviour noted in the hot leg loop seals would yield information on *flooding* or the *carry-up* of water by steam flowing through the loop seal. Finally, the experiments would provide information on *liquid hold-up* in the steam generator and *countercurrent flow* out of the steam generator.

In 1985 the Technical Research Centre of Finland (VTT) and the Lappeenranta University of Technology performed a series of natural circulation experiments in the REWET-III (Hongisto 1986). No longer running, REWET-III was a small-scale integral test loop with a volumetric scaling ratio of 1 to 2333. Like PACTEL, REWET-III modelled a VVER-440 reactor loop with a horizontal steam generator. Running at a maximum core power of 90 kW, which represents about 15% of full reactor power, REWET-III addressed low pressure thermal hydraulics (maximum 10 bar). Comparing the results of the SIR experiments in the PACTEL with the 1985 natural circulation experiments performed in REWET-III provides useful information on the effects of relative scaling and further confirms the REWET-III data. This was the first objective mentioned in above.

The experiment procedure for the SIR-23, SIR-20, and SIR-21 experiments was straightforward. In each case, the plant operators brought Loop III of the PACTEL to the planned initial conditions with the primary-coolant pump running. The initial conditions included primary- and secondary-side pressure, primary- and secondary-side full liquid levels, and core power. Valves isolated loops I and II. PACTEL loop III stabilised at the initial conditions for about one hour.

Each experiment began when the operators started the data acquisition system to record initial conditions for 1000 seconds. Next, the operators shut off the loop III pump and allowed the primary coolant to settle into steady state natural circulation. At nearly 1000 seconds from pump shutdown, the operators isolated the pressuriser. At precisely 2000 seconds from the start of data logging, they drained roughly 40 kg of coolant from the loop and allowed the loop to stabilise again. Primary-coolant draining continued in increments of about 40 kg every 1000 seconds except for one step in the SIR-23 experiment. In that step, to avoid exceeding the loop maximum pressure, the operators began draining early. Table 2 shows the initial conditions for each experiment.

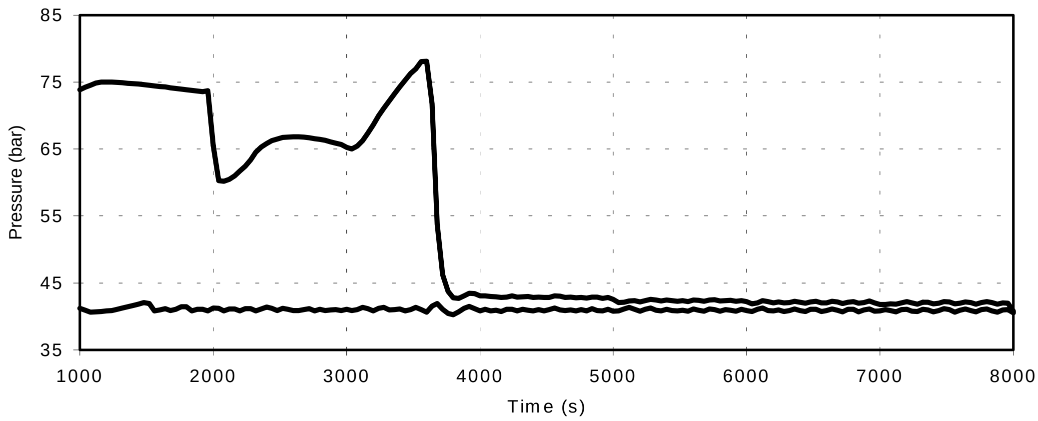
Table 2. Prescribed initial conditions for SIR-23, SIR-20, and SIR-21 experiments.

	SIR-23	SIR-20	SIR-21
Primary pressure (bar)	**75**	**40**	**16**
Secondary pressure (bar)	**42**	**12**	**3**
Core power (kW)	**115**	**115**	**115**

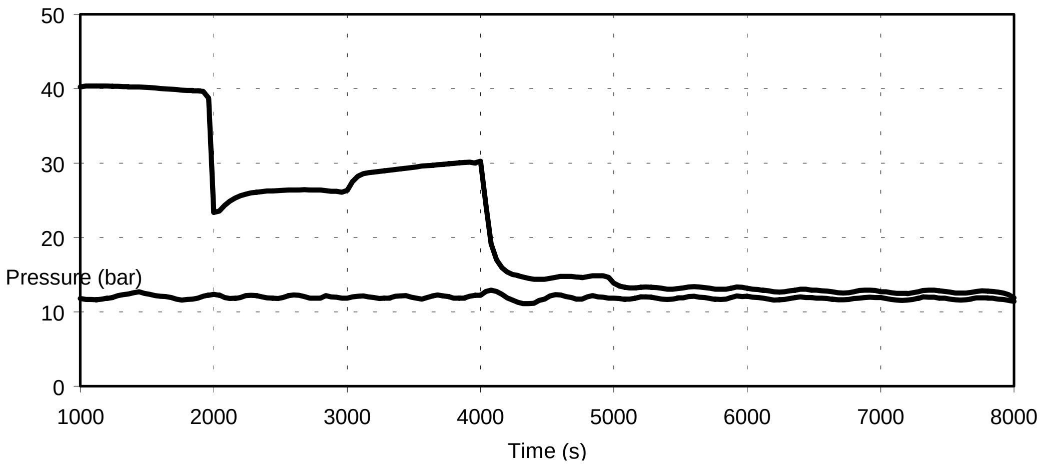
Fig. 31 presents the history of primary loop coolant pressure measured in the upper plenum and secondary-side pressure for each of the SIR-23, SIR-20, and SIR-21 experiments. Both pressures were important features in the natural circulation behaviours. Secondary-side pressure remained constant in all three experiments. Secondary-side pressure was important because it controlled the temperature of the coolant in the cold leg region, which in turn fixed the size of the buoyancy forces. Primary-side pressure was important because it determined the water swell level above the core. This affected the timing of the flow regime changes and the speed of the steam flowing through the hot leg.

Fig. 32 presents the history of downcomer mass flow rate for each of the SIR-23, SIR-20, and SIR-21 experiments. For mass flow rate, the principal difference from experiment-to-experiment was the coolant's nature. Primary- and secondary-side pressures, with core power, determined this nature. Within each experiment, the nature of the coolant changed according to the primary-side pressure and according to changes in coolant inventory. As inventory dropped, the water above the core and in the hot leg changed from subcooled to saturated, from saturated to two-phase, and finally from two-phase to steam. These transformations affected both the frictional forces and the buoyancy forces. The region in the loop that most reveals the changing nature of the coolant is the inlet side of the hot leg loop seal. Therefore, besides the mass flow rate histories for each experiment, the graphs in Fig. 32 below show the liquid level measurements for the inlet side of the hot leg. The liquid level is the grey line. The black data plot in each graph is the mass flow rate history.

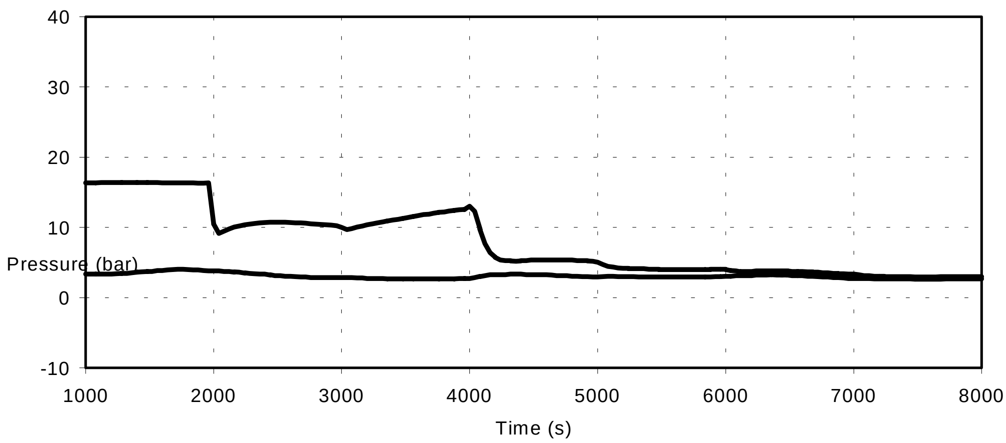
Fulfilling the first and third objectives of these experiments is self-evident. About the first objective, similar behaviours existed in both the previously performed experiments and the SIR experiments. About the second objective, the computer code predictions for these experiments using the RELAP5/Mod3.1 code predicted the main thermal-hydraulic. The computer model did not predict the chaotic behaviours such as wave instability leading to reflooding or the rapid variations in mass flow because of other local interactions between steam and water. Also, the computer model did not precisely pinpoint the transitions in natural circulation modes.



SIR-23

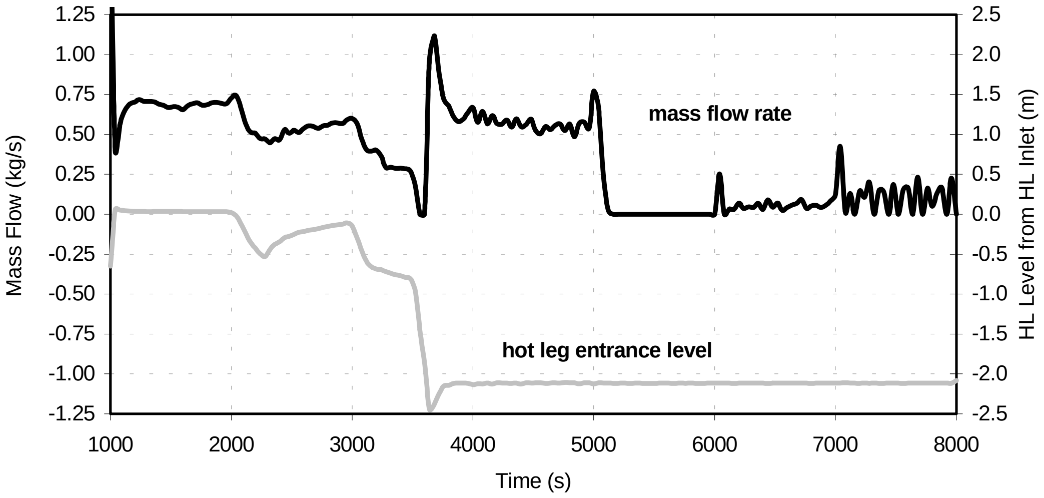


SIR-20

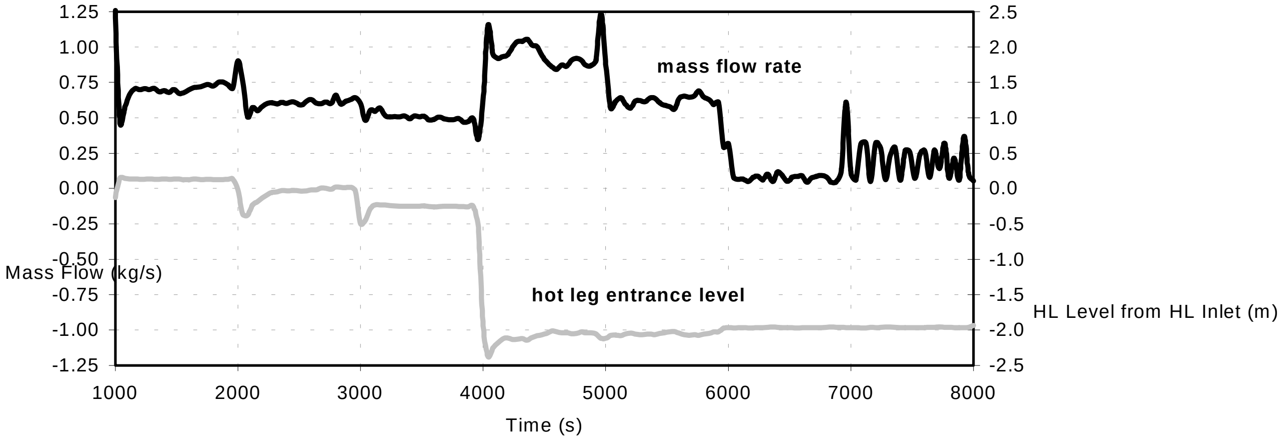


SIR-21

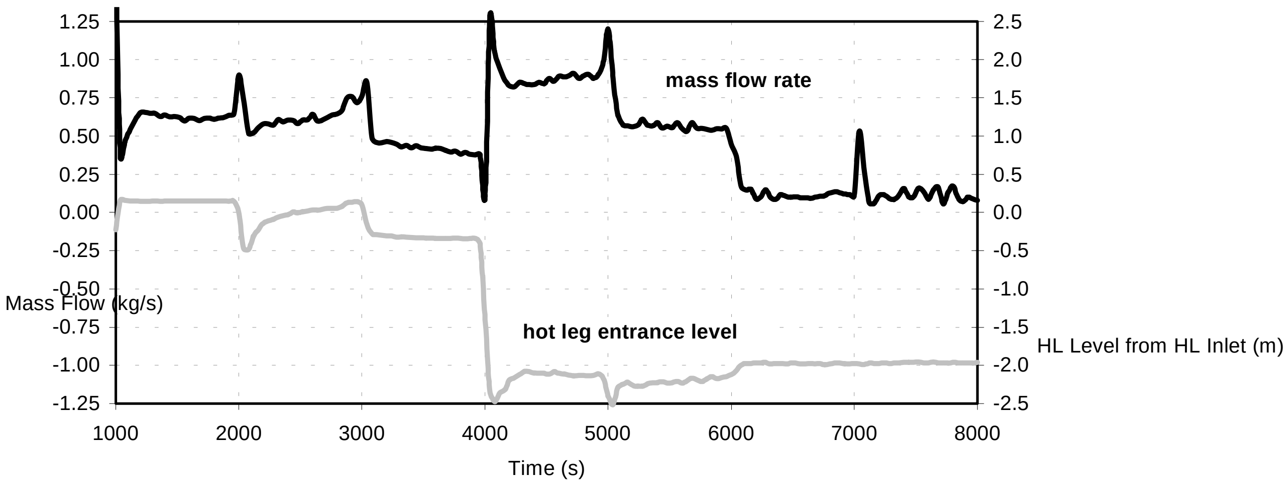
Figure 31. Primary and secondary pressure for SIR experiments.



SIR 23



SIR 20



SIR 21

Figure 32. Downcomer liquid mass flow rate for SIR experiments.

The second objective addressed the influence of primary-side pressure on natural circulation behaviour. Flow stagnation occurred only in the high pressure SIR-23 experiment where the primary pressure was higher and the resulting boiling swell levels were less. Also, clearing of the hot leg was most difficult at the higher pressure. Finally, in the higher pressure SIR-23 experiment, the measured mass flow rate during natural circulation with a two-phase water and steam mixture in the hot leg was lower than in the two lower pressure experiments. The timing of the transitions from one natural circulation mode also depended on primary-side pressure level.

The fourth objective regarded the new steam generator design. The new steam generators behaved similarly to the previous ones in terms of the natural circulation behaviours. Besides the primary experiment objectives, information on other behaviours of interest resulted from the SIR experiments. The experiments revealed some important hot leg loop seal behaviours during inventory loss. Both countercurrent flow and reflooding occurred. There was chaotic flow behaviour at low inventories because of liquid held up in the vertical section of the hot leg leading into the steam generator. In the SIR-23 experiment, flow stagnation occurred. There was not enough instrumentation in the hot leg to evaluate countercurrent flow rigorously. There were not enough instruments to clarify the related reflood behaviours or the void fraction distributions. New experiments with added instrumentation would provide a more detailed picture. Also, data showed that natural circulation depended on core power level, that core mass flow rate depended on primary-coolant inventory, and that core mass flow rate depended on flow regime.

3.3.2 Small break loss of coolant experiments (SBL-30, SBL-31, SBL-32 and SBL-33)

The goal of the small break LOCA experiment series was to study the behaviour of the new taller steam generator design and new accumulators of PACTEL. The experiments also focused on the mechanisms of a possible inherent boron dilution transient during SBLOCA’s in VVER-440 geometry. Especially of interest was the period when the hot leg loop seals were open and a boiler condenser heat transfer mode prevailed. A quantitative objective was to get an estimate of boron free condensate formed in the steam generators during the transient. PACTEL operators used secondary side feed and bleed as an operator action and accident management procedure in three experiments.

PACTEL operators carried out a series of four SBLOCA experiments, as tabulated in Table 3, in November and December 1995. The high pressure safety injection system, and both, downcomer and upper plenum accumulators, were all in use for the first time in PACTEL experiments. In all the tests, the break located at the bottom of the Loop 2 cold leg near the downcomer. The quick look report includes an overview of the results (Puustinen 1996).

Table 3. Series of SBLOCA experiments

RUN NO.	BREAK SIZE	OBJECTIVES AND CONDITIONS FOR THE EXPERIMENTS
SBL-30	∅ 1.0 mm	comparison test for SBL-7, behavior of new steam generators, pressurizer isolated
SBL-31	∅ 2.5 mm	testing of accumulator performance and secondary feeding and bleeding procedure
SBL-32	∅ 2.8 mm	boron dilution mechanism, accumulators, HPIS and secondary feeding and bleeding as an operator action
SBL-33	∅ 3.5 mm	boron dilution mechanism, accumulators, HPIS and secondary feeding and bleeding as an operator action

The main goal of the first experiment of the series (SBL-30) was to study affects of new steam generator design on primary system behaviour in SBLOCA conditions. The initial and boundary conditions of the experiment SBL-30 match those of experiment SBL-07, which had the old steam generator design. According to the SBL-30 experiment the effect of the new taller steam generator design on the behaviour of natural circulation was minor. The general course of the experiment was similar to the reference experiment. However, the SBL-30 experiment included more hot leg loop seal clearings and refillings than SBL-07. The main reason was the slight inclination of the new steam generator tube bundle. It helped the flow of condensate, collected in the steam generator tubes, back to the hot collector and to the up-flow leg of the hot leg loop seal. The better internal circulation caused by the taller tube bundle and the shorter heat exchange tubes also contributed to this effect.

The experiments verified the sensitivity of the boron dilution mechanism to break size, as predicted by computer code simulations of the Loviisa VVER-440 reactors (Kantee 1994). With the ∅2.8 mm break size (~1 % leakage in the reference reactor), the period when boron dilution could have happened was too short to be significant. With the ∅3.5 mm break size (~1.5 % leakage in the reference reactor), the conditions in the loop achieved those needed to form a significant unborated water slug. For about 2500 seconds, the primary pressure was above the LPIS head (7.5 bar), and the core water-level remained below the hot leg entrance elevation (Fig. 33). Besides this, all the hot leg loop seals cleared almost simultaneously (Fig. 34) and enabled the boiler condenser heat transfer mode, a prerequisite for boron dilution, in all three steam generators.

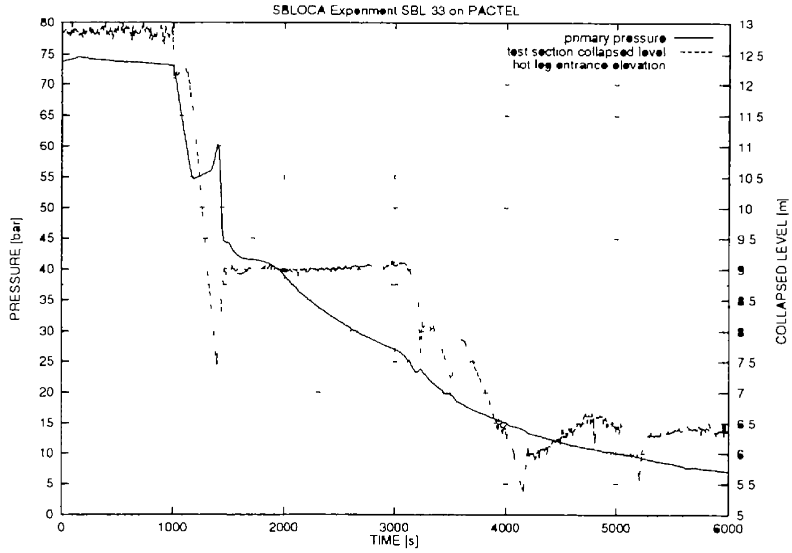


Figure 33. Primary pressure and core water level in SBLOCA experiment SBL-33.

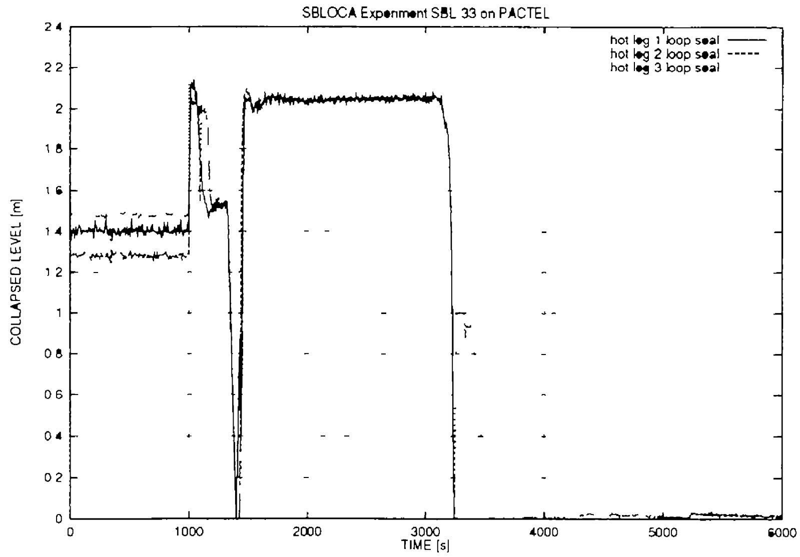


Figure 34. Loop seal water levels in SBLOCA experiment SBL-33.

3.3.3 Primary secondary leakage experiments (PSL-05, PSL-06 and PSL-07)

The first series of PACTEL steam generator multiple tube rupture experiments (PRISE) were run in June 1994 within the PROPA project. It included three experiments, each with different recovery actions. PACTEL (Parallel Channel Test Loop) is a full height, medium-scale integral test facility (volumetrically scaled 1:305) designed to simulate the thermal-hydraulic phenomena characteristic of the Finnish Loviisa PWR. VTT Energy together with the Lappeenranta University of Technology run the facility. PACTEL has three primary coolant loops with pressurizer, primary coolant pumps and horizontal steam generators, high pressure emergency core cooling system (ECCS), and low pressure ECCS with two accumulators. The peak operating pressures in the primary and secondary sides are 8 MPa and 4.6 MPa, respectively. The reactor vessel is simulated with a U-tube construction consisting of separate downcomer and core sections. The core comprises of 144 full length, electrically heated fuel rod simulators with a heated length of 2.42 meters. The maximum total core output is 1 MW, or 22 % of the scaled full power. The three coolant loops with double capacity steam generators model the six loops of the reference power plant. Each steam generator has 118 U-tubes with an average length of 2.8 m. The experiments focused on the possibility of leak flow reversal occurring and a partially diluted or completely unborated water plug forming in the primary side because of operator intervention. The results showed that the reversal is possible in suitable conditions. Because the first series utilized the old facility configuration (low steam generators and only one accumulator), the PACTEL operators performed a new set of experiments in February 1996. The new experiments used all three steam generators and two accumulators. The main goal was to find the worst possible conditions during which the leak flow reversal can occur.

The new PRISE experiments were based on the current regulations for operator actions during a state of emergency in the Loviisa plant. PACTEL operators used secondary-side feed and bleed as an accident management measure in all experiments. The first experiment (PSL05) also included primary bleed and feed. The operators of PACTEL started primary bleed and feed by opening the pressurizer relief valve and by using the high pressure injection system (HPIS). The second experiment (PSL06) was similar to the first one but with an assumption of total failure of the HPIS system. In the last experiment (PSL07), the operators did not use primary bleed and feed.

All experiments used all three steam generators of PACTEL; two intact and one broken steam generator. The experiment conditions assumed the main isolation valves had stuck open, which prevented isolating the broken loop. The leak flow reversal occurred in the first two experiments where the operators used primary bleed (Fig. 35 and 36). In the last experiment, the pressurizer spray and the cooling of the primary side by the intact steam generators were not enough to reverse the leak flow significantly. The pressure difference between the primary side and the secondary side of the broken steam generator was so small that no notable decrease of the secondary side collapsed level was visible (Fig. 37).

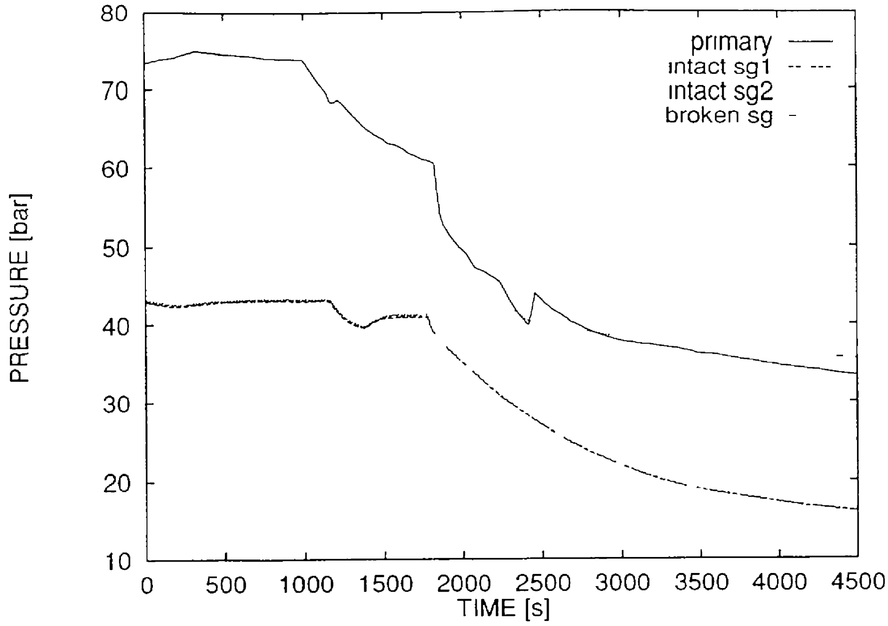


Figure 35. Primary and secondary pressure in PSL05.

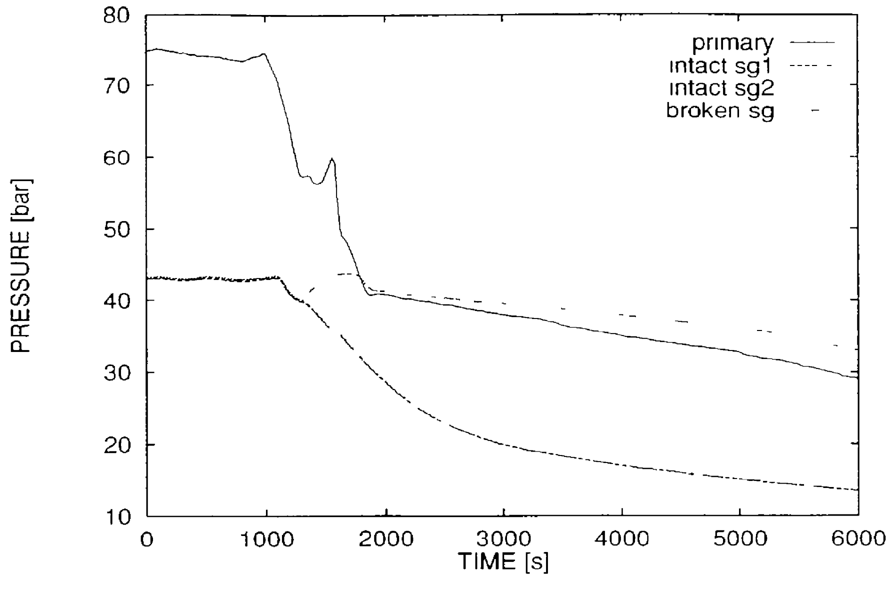


Figure 36. Primary and secondary pressure in PSL06.

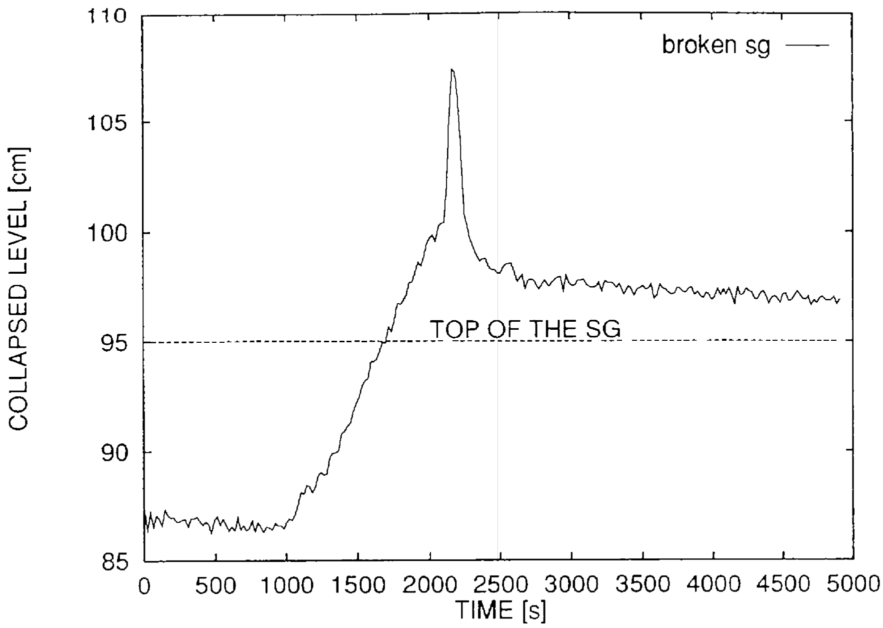


Figure 37. Broken steam generator collapsed level in PSL07. Levels higher than 95 cm are measured from steam collector.

The PRISE experiments showed the leak flow reversal was significant only when the operators actively depressurized the primary side by, for example, primary bleed. The quick look report gives an overview of the results of the experiments (Riikonen & Semken 1996). The experiment data report gives more detailed analyses of the experiments (Riikonen 1996b).

3.3.4 Passive safety injection experiments (GDE-21 through GDE-35)

PACTEL operators ran two series of five experiments each within the PAHKO project to investigate passive safety injection system performance during small break loss-of coolant accidents. The passive safety injection system consisted of a Core Make-up Tank (CMT), a Pressure Balancing Line (PBL), and an Injection Line (IL), Fig. 38. The PBL connected the CMT to one cold leg. The IL connected it to the downcomer. The purpose of this passive safety injection system is to provide high pressure safety injection water to the primary circuit of a nuclear power plant during loss-of coolant accidents. The CMT is at the same pressure as the primary circuit but the isolation valve in the IL is closed. The operators open the IL isolation valve in accident situations, for example from a low pressurizer level signal. The CMT and the IL are initially full of cold water. PACTEL operators filled the PBL with hot water to provide the necessary density difference for natural circulation. Natural circulation began when the operators opened the IL valve.

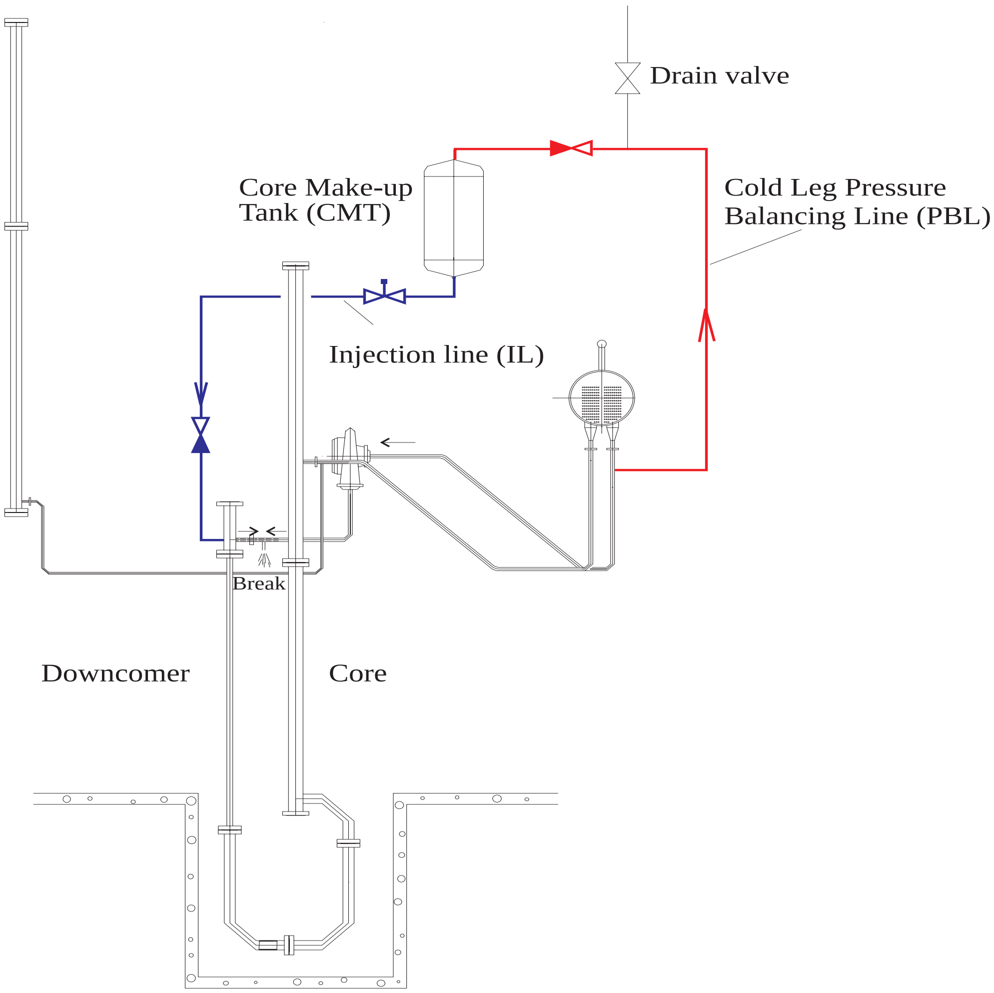


Figure 38. General view of the PACTEL facility and the passive safety injection system.

The passive safety injection system studied is similar to the one designed for the Westinghouse AP-600 reactor concept. The rest of the primary circuit of PACTEL differs from the AP-600 design. The first series of passive safety injection experiments focused on the influence of the break size on the passive safety injection system performance. The second series studied the influences of break location on CMT behaviour. The analyses of the first experiment series has been completed and is available. The experimenters completed the second series in the middle of December 1996. The experiments are a part of the European Commission Nuclear Fission Safety Programme. Before the experimental work began the partners of the project made a review of existing experimental and analytical work around passive safety injection systems of APWR's (Tuunanen et al. 1996). PACTEL operators measured the flow resistance of the CMT lines in a separate experiment (Semken 1996a and 1996b). This data is an important boundary condition in the computer code simulations of the experiments.

Fig. 39 presents the flow rates in the PBL and IL for experiment GDE-24 (3,5 mm cold leg break). The first phase of CMT operation started when the operators opened the IL valve. During this first phase (recirculation phase), hot water flowed to the CMT, replacing cold water there. This phase continued as long as the water-level in the cold leg is above the PBL connection position. The second phase (oscillation phase) started when the cold leg level dropped to the PBL connection position and two phase mixture started to flow to the CMT. During this phase, mass flow in the CMT lines oscillated, partly because of condensation of steam at the top of CMT. The third phase (injection phase) started when the level in the cold leg dropped below the PBL connection position. The CMT injection mass flow rate increased during this phase, due to an increased density difference between the PBL and IL.

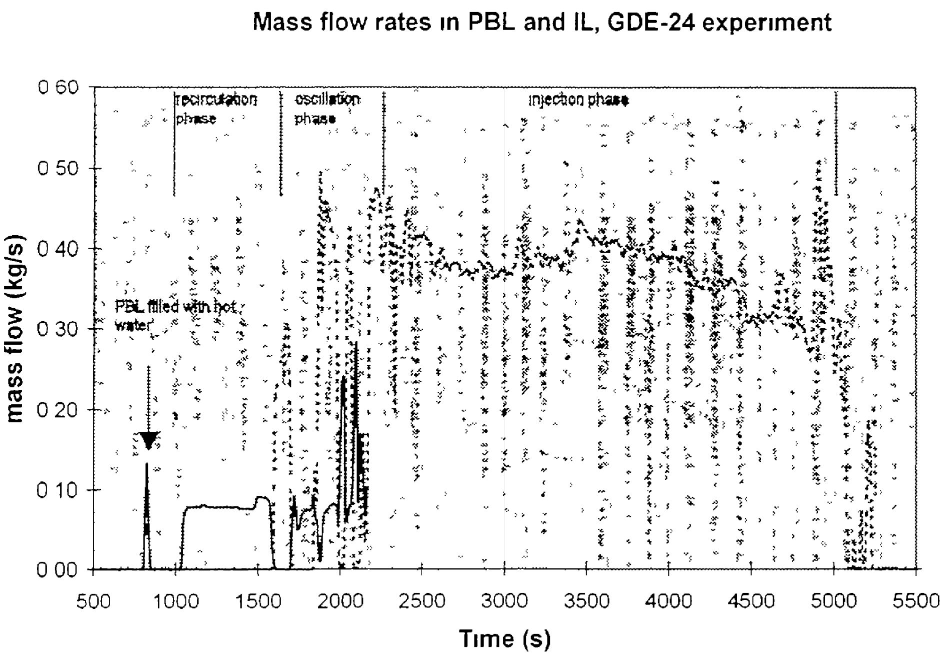


Figure 39. Mass flow rate in the PBL and IL (GDE-24 experiment).

Fig. 40 presents the conditions in the CMT during the injection phase. The top of the CMT was full of steam. A layer of hot water separated steam from cold water. The hot water layer consisted of two parts. The upper part of the layer was close to saturation conditions. Thermally-stratified liquid formed the lower part. In the CMT, condensation of steam to the walls and possibly to the liquid occurs. Cold CMT walls cooled the hot water layer. During the depressurization phase of SBLOCA, flashing of hot liquid was also possible. Since a stable hot liquid layer between steam and cold water existed in the CMT, the condensation in the CMT remains at a reasonably low-level. If the hot liquid layer breaks down (possibly by the flow of water from the PBL or through flashing), rapid condensation can stop the safety injection from the CMT.

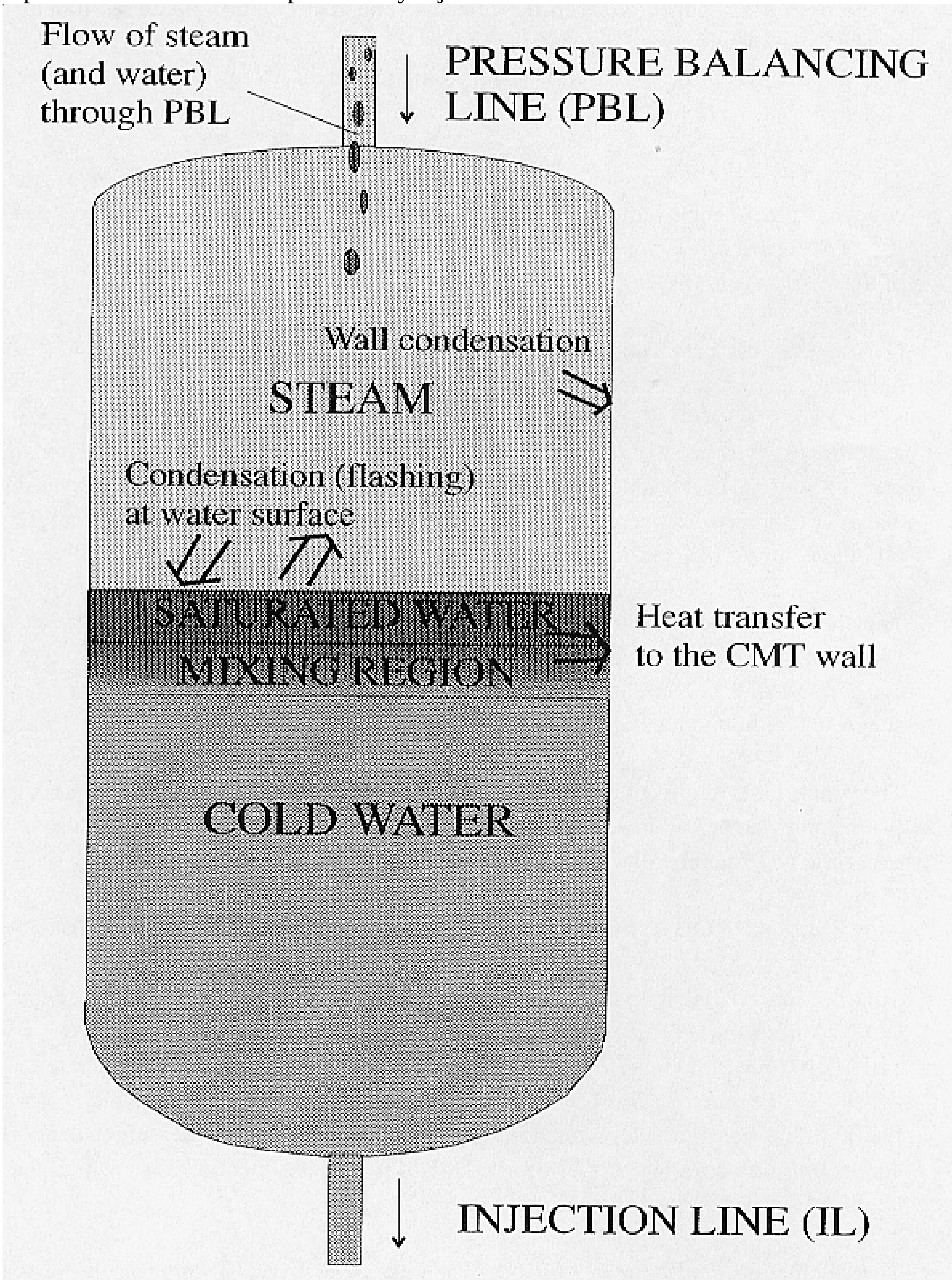


Figure 40. Conditions in the CMT during injection phase.

The break size effects on the CMT behaviour can be summarised as following:

- the recirculation phase is longer, and the resulting hot liquid layer, thicker in the experiments with smaller break size;
- the oscillation phase between the injection and recirculation phases is longer in the experiments with small break size; and
- the local heat flux to the CMT wall is higher in the experiments with larger break size.

The CMT worked as planned in all experiments of the first series. Problems with rapid condensation occurred in one experiment of the second series. In this experiment, PACTEL operators removed the flow distributor (sparger) from the CMT. The purpose of the sparger is to divide flow from the PBL to the CMT horizontally.

During the test preparation period, it was observed that heating of the PBL was necessary to start up the recirculation flow. To further study the influence of the heating of the PBL, the PACTEL operators will run an experiment without PBL heating. Further experiments with small break sizes are also necessary. If the break size is small, it is possible the CMT becomes full of hot water during the recirculation phase, and the density difference between the PBL and the injection line disappears. This might cause problems during the start up of the injection phase of CMT operation.

The analysis of the test data supports the suggestion of Westinghouse for application of McAdams correlation for the heat transfer from hot liquid to the CMT wall. The use of the Nusselt film condensation correlation for condensation onto the CMT walls also is suggested, although the correlation gives high values for the heat transfer coefficient.

The quick look report gives an overview of the experimental results of the first series (Tuunanen 1996a). An experimental data report gives a detailed analyses of the experiments (Tuunanen 1996b).

3.3.5 APROS simulations of passive safety injection experiments

This section describes the calculation results for PACTEL gravity-driven experiment GDE-24 from the APROS 4.02 computer code. The simulated sequence was a cold leg SBLOCA (3,5 mm break) with ECC water from the passive safety injection system (CMT, see Chapter 3.3.3). The CMT is at a higher elevation than the main part of the facility. The passive safety injection system includes also a pressure balancing line connecting one cold leg to the top of the CMT, and an injection line, which leads the cold CMT water to the downcomer.

The calculation model for the APROS code uses the PACTEL input deck prepared for the small break LOCA analyses (Plit 1996). The new CMT model includes 30 equal length nodes. The dense nodalization is necessary for modelling of the expected thermal stratification in the CMT. The flow areas of the top and bottom nodes are smaller than in the middle to get correct volume for the round ended tank. The PBL and the IL loss coefficients agree with the measured values (Semken 1996a).

The first 1000 seconds of the experiment were a steady state run with primary pumps on and pressurizer heaters controlling the primary pressure. At the end of this period, the simulation included filling the PBL with warm primary water to start the natural circulation through PBL to CMT. APROS simulated this part of the sequence reasonably well. The transient started by opening the break simulation valve simultaneously with the pump stop signal. About 30 seconds later the low pressurizer level signal led to opening the IL valve and switching off the pressurizer heaters.

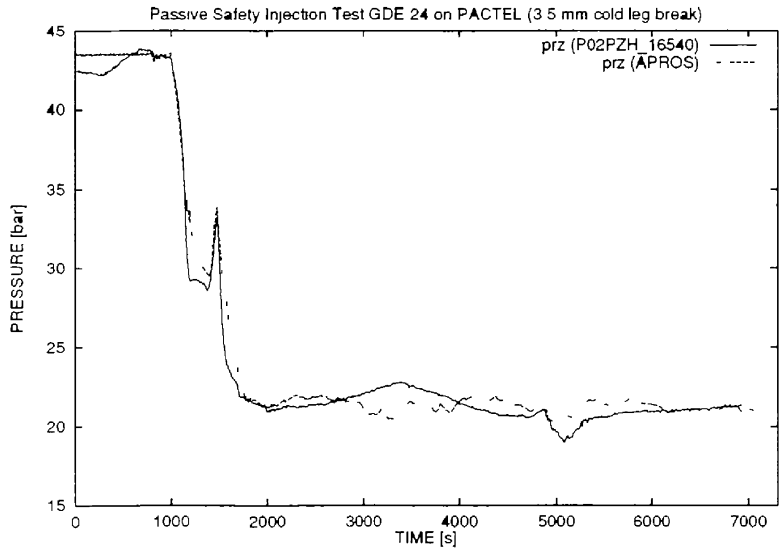


Figure 41. Primary pressure in APROS simulations of GDE-24 experiment.

The APROS code simulated accurately the first part of the pressure decrease towards the saturation after the break opening. However, during the second part, the pressure dropped more in the experiment than in the simulation, Fig. 41. Decreasing of the time step improved the simulation of the pressure transient, but could not solve the problem. The code simulated well the pressure peak occurring when upper plenum water-level reached the hot leg elevation. APROS simulated well the mass flow rate during the recirculation phase, during which hot water from the cold leg flows to the top of CMT, but the phase lasted longer than in the experiment, Fig 42. The main problem in the simulations was the oscillations of the CMT injection flow rate. This affected the behaviour of the whole primary system. In the simulations, the code was not able to calculate accurately thermal stratification in the CMT. The uppermost water layer remained subcooled, which led to condensation and stopped injection from the CMT. The water layer reached saturation and injection started only when the node at the water surface was almost empty. The results improved if the analysts increased the number of nodes in the

CMT. This did not, however, remove mass flow rate oscillations. Because of the problems with condensation in the CMT, the core heat up at the end of the experiment was slightly delayed.

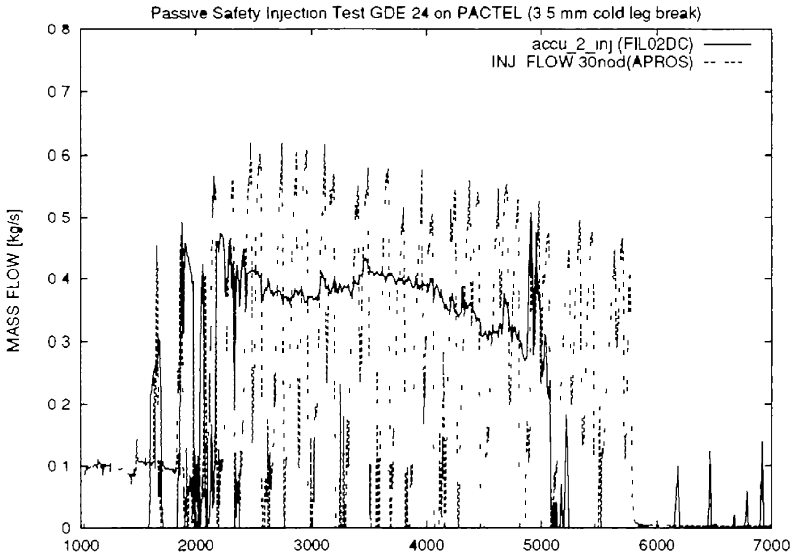


Figure 42. CMT injection flow rate in APROS simulations of GDE-24 experiment.

3.3.6 RELAP simulations of PRISE experiments

PACTEL operators performed the first series of PACTEL steam generator multiple tube rupture experiments in June 1994. The series included three experiments with different recovery actions. The experiments focused on the possibility of a leak flow reversal occurring and a partially diluted or completely unborated water plug forming in the primary side due to operator intervention.

Two experiments (PSL02 and PSL04) were simulated with RELAP5/Mod3.2. The first experiment (PSL02) was based on the current regulations for operator actions during a state of emergency in the Loviisa nuclear power plant. The second experiment (PSL04) assumed a total failure of the HPIS and spray actuation. PACTEL operators used secondary feed and bleed at the intact steam generator as a recovery action in both experiments. The experiments used two of the three available steam generators of PACTEL; one in the intact and one in the broken loop. The main isolation valves were in the stuck-open position which prevented the broken loop isolation.

The results of the simulations agreed satisfactorily with the experiments. Timing of some events and the behaviour of the steam generators were the main differences. The main reason for the timing problem was the broken steam generator behaviour in the simulation. Level rise in the broken steam generator influenced events around the loop and the delay of the level rise simulation affected the whole transient. The simulation included sensitivity studies to fine-tune the behaviour of the steam generators. For example, the analysts changed the parameters of the separator component, varied the size and distribution of the heat losses, and changed the parameters of the leak. These changes did not have any significant effects on the steam generator behaviour.

The calculations included strong pressure oscillation in PSL04 (Fig. 43). There were also accumulator injection modelling difficulties. The analysts could not find any reasonable cause for the oscillations. Unrealistic and high pressure loss coefficients of the accumulator injection lines were necessary to achieve the correct mass flow rate of the accumulator injection.

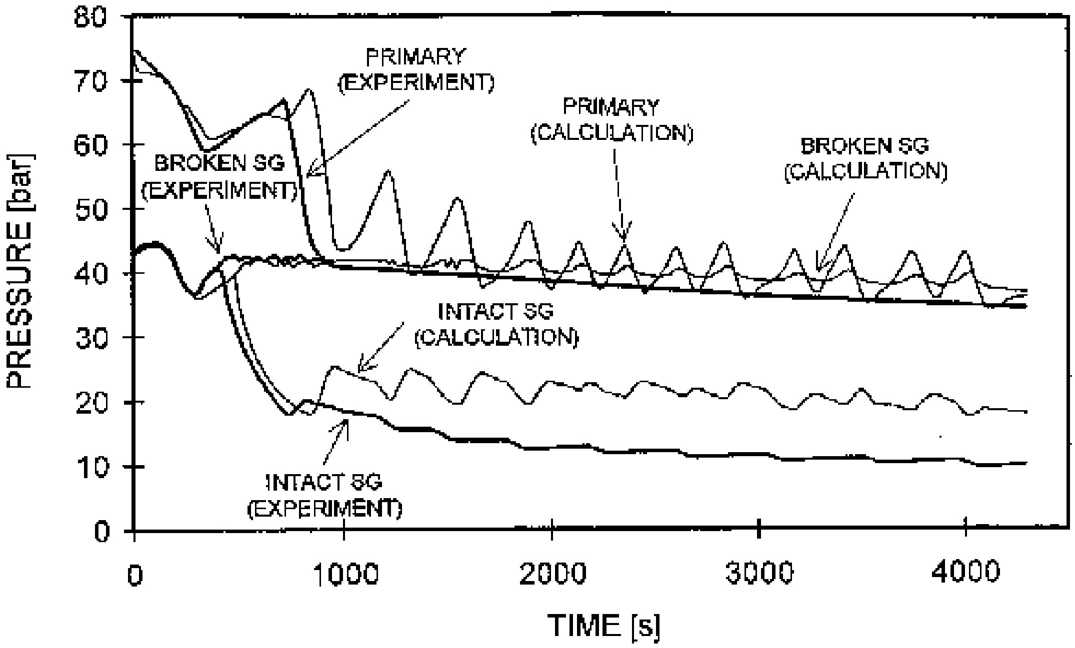


Figure 43. Measured and calculated primary and secondary pressures in PSL04.

In the experiments, the pressure difference between the primary-side and the secondary-side of the broken steam generator was too small for a reasonable estimation of the flow rate or even the direction of the flow. In the PSL04 experiment, the primary pressure oscillation also caused oscillation in the break flow. The average flow was, however, close to zero. In PSL02, the leak flow reversed only temporarily when the pressurizer spray started at the end of the experiment. This was expected because the primary pressure collapsed, and the broken steam generator pressure did not follow the primary pressure.

3.3.7 RELAP5 simulations of small pressure natural circulation experiments

In preparation for the SIR-23, SIR-20, and SIR-21 experiments, analysts prepared and ran five computer simulations with the RELAP5/Mod3.1 code (Semken & Tuunanen 1996). The results of these simulations enabled experiment planners to select the best operating features for the three experiments. The input model for the SIR experiment RELAP5/Mod3.1 code came from a previously developed input model used in the post-experiment analysis of PACTEL small break LOCA experiment SBL-22 (Riikonen et al. 1995). For the SIR experiment computer simulations, the analysts altered the input model by removing the cold leg break and adding a model for draining primary coolant from the lower plenum region. Also, they isolated the first and second primary loops so the model would represent the third loop of the PACTEL.

The RELAP5/Mod3.1 code simulated primary-coolant draining in steps during natural circulation. The main operating features selected for the simulations were primary pressure, secondary-side pressure, and coolant drained with each step. Core power was 115 kW, which represented 2.6 % of full power for the reference reactor. Table 4 summarises the main operating features for the five simulations. Each simulation began with steady state forced circulation of the primary coolant. This continued for 1000 seconds. At 2000 seconds into the simulation, the pressuriser isolated, and the first draining step began. The next four draining steps occurred at 1000-second intervals.

Table 4. Parameters of RELAP5/Mod3.1 simulations for SIR-23, SIR-20, and SIR-21

	Sim 1	Sim 2	Sim 3	Sim 4	Sim 5
Primary pressure (bar)	**75**	**60**	**40**	**25**	**7**
Secondary pressure (bar)	**42**	**20**	**12**	**6**	**1**
Drained mass (kg/drainage)	**36**	**39**	**40**	**42**	**44**

The following graphs 44 and 45 present the primary-coolant mass flow rates predicted by the RELAP5/Mod3.1 code simulations. Graph 44 shows steady state natural circulation flow rates at various coolant inventories for each simulations. Graph 45 show the mass flow rate histories. A third graph, 46, shows how primary pressure varied during the simulations.

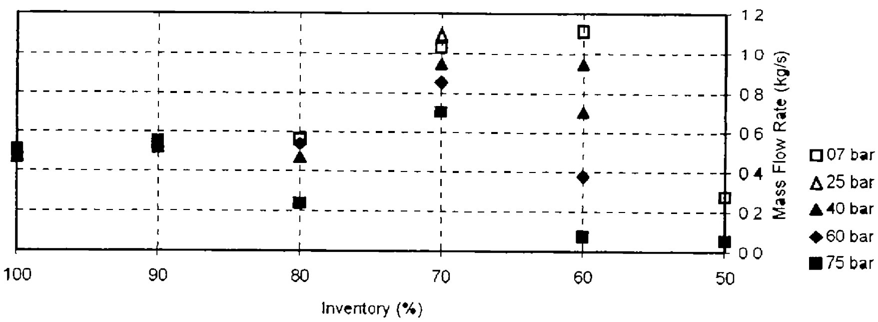


Figure 44. Mass flow versus inventory for SIR simulations.

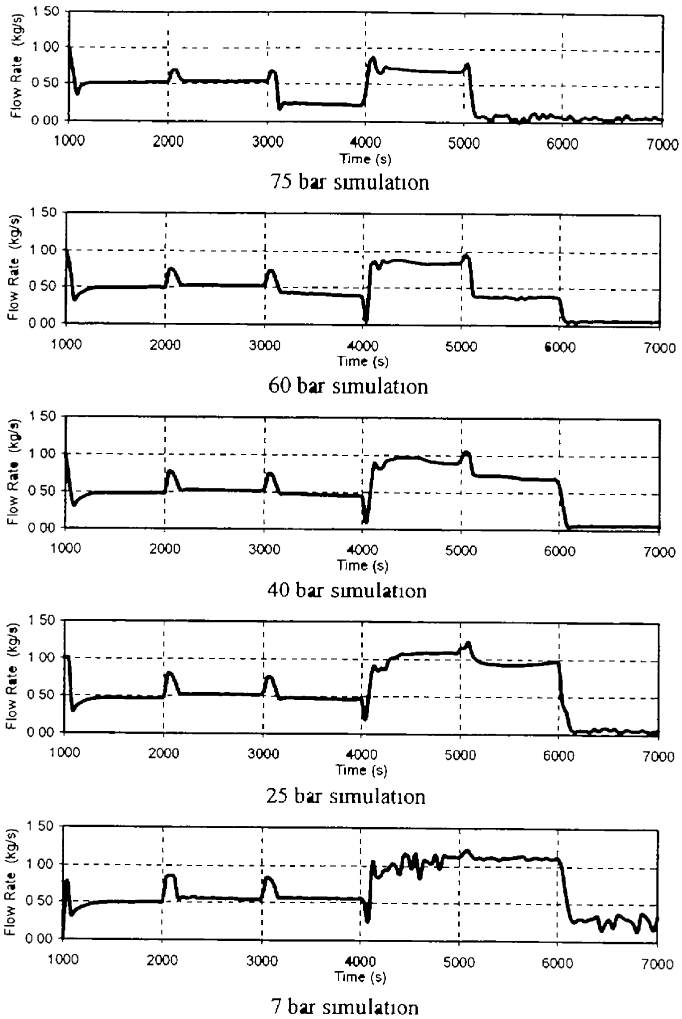


Figure 45. Mass flow rate in different RELAP5 simulations.

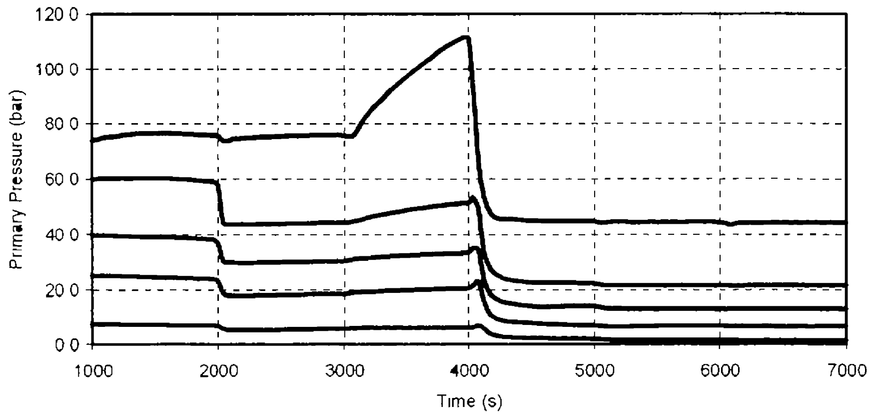


Figure 46. Primary side pressure histories for SIR simulations.

Qualitatively speaking, the code simulations showed that primary pressure level is an important variable. It controls the densities in the loop, the behaviour of the *swell level* above the core, and the void fraction profile in the hot leg. Primary pressure level also controls the speed of steam flow during the boiler-condenser or reflux condensation natural circulation modes. Considering the RELAP5/Mod3.1 code simulations, natural circulation in one loop of the PACTEL should conform to the following statements.

- When primary pressure is lower, saturated one-phase liquid mass flow rate during natural circulation is lower.
- When primary pressure is lower, two-phase mixture mass flow rate during natural circulation is higher.
- When primary pressure is lower, flow stagnation and the transition to two-phase mixture natural circulation occurs at lower inventories.
- When primary pressure is lower, the transition to boiler-condenser natural circulation occurs at lower inventories.

3.3.8 Objectives for the years 1997 and 1998

During the years 1997 and 1998 TEKOJA experiments will continue investigations of basic thermal hydraulics of VVER-type PWRs. The main objective is to simulate ATWS situations and natural circulation phenomena with non-condensable gas in the primary circuit. A group of experts made a proposal for experiments to simulate the ATWS situation (Tuunanen 1996c). The group suggested starting experimentation with two basic experiment series. The first series should simulate a maximum pressure scenario in control rod withdrawal situations. The second series should investigate the start-up of natural circulation by, for example, charging water from the pressurizer. In this second series, the PACTEL operators will start the experiment from boiler condenser mode natural circulation conditions, from a situation where the pressurizer is full of water. Using available operator actions, the operators should lower primary pressure to make water from the pressurizer available for core cooling. These two experiment series do not require large changes to the PACTEL facility. The experiments belong to the test program of the year 1997. A third series of ATWS experiments is part of the year 1998 test program. In this series a feed-back system from core fluid temperature and void fraction to the core power should be in use. This series requires changes to the loop operation and control systems. The 1997 test program also includes a small break LOCA experiment simulating possible water-hammer phenomena in the downcomer. In one passive safety injection experiment (GDE-23, 5 mm cold leg break), there were problems with condensation-induced water-hammer in the downcomer. The RETU project group proposed to repeat this experiment with normal accumulators, to observe if a similar phenomena could occur.

The main objective of the PAHKO project for the years 1997 - 1998 is to complete the experimental series, analyses and documentation, as specified in the contract with the

European Commission. The second series of experiments was completed in December 1996, and the third one will be carried out in Autumn 1997. The analyses of the second series was started in January 1997. The main objective of the second series was to investigate the affects of break location in the CMT behaviour. This series used a smaller CMT. In the third series, the PACTEL operators will move the CMT to a higher position to increase the driving head for ECC injection. Experiments of this series will study also the influences of different operator actions on CMT behaviour. Analysts will simulate selected experiments from both series with APROS code.

Table 5 summarises the TEKOJA and PAHKO experiment programs for the years 1997 and 1998. Besides those experiments presented in the Table 5, the project group continues experiments to investigate the influences of specific features of the VVER reactor (hot leg loop seals and horizontal steam generators) on natural circulation characteristics. Completing documentation of the earlier experiments will also require resources.

Table 5. Preliminary test program for publicly funded experiments.

PRELIMINARY SCHEDULE OF PUBLICLY FUNDED PACTEL EXPERIMENTS IN 1997 AND 1998				
TEKOJA				
1997	Schedule	Number of experiments	Phenomena investigated	Comments
Start-up of natural circulation by charging pressurizer water	2/97	~3	Natural circulation Accident management measures	
5 mm cold leg SBLOCA	3/97	1	Condensation phenomena in downcomer	
Control rod withdrawal	10/97	~3	Maximum pressure scenario	
1998				
Natural circulation experiments with non condensables	2/98	~5	Non-condensable effects on natural circulation	Nitrogen injection & measurement system needed
Steam generator collector break	5/98	1	Accident management measures	Process control system modifications needed
ATWS experiments	8/98	~3	Core feed back effects on natural circulation phenomena	Process control system modifications needed
PAHKO				
(1996) 1997	Schedule	Number of experiments	Phenomena investigated	Comments
Third series with CMT at high elevation	8/97-9/97	5	influences of operator actions and increased driving head on the CMT behaviour	

References for section 3.3

Bánáti J. 1995. Assessment of RELAP5/MOD3.1 against the 4th IAEA standard problem exercise performed on PMK-2 facility, VTT Energy, TEKOJA 6/95.

Hongisto, O. 1986. A research on decay heat removal by natural circulation using an experimental facility for the modelling of VVER-440 pressurized water reactor (in Finnish). Diploma Thesis. LTKK. (Luonnonkierrolla tapahtuvaan jälkilämmönpoistoon liittyvä tutkimus VVER-440 painevesireaktoria simuloivalla koelaitteistolla.)

Kantee, H. 1994. Boron dilution. Analysis of small break (in Finnish). IVO International Oy. Analysis report LO1-K859-0130. (Boorin laimeneminen. Pienen vuodon analysointi.)

Plit, H. 1996. Validation of a simulation software using a test facility for the description of a nuclear power plant (in Finnish). Diploma Thesis. (Simulointiohjelmiston kelpoistaminen ydinvoimalaitosta kuvaavalla koelaitteistolla.) Helsinki University of Technology.

Puustinen, M. 1996. Small break loss-of-coolant accident experiments SBL-30, SBL-31, SBL-32 and SBL-33 in the PACTEL facility. Quick look report. VTT Energy, TEKOJA 6/96.

Riikonen, V. & Semken, S. 1996. Steam generator multiple tube rupture experiments (PSL05, PSL06 and PSL07) on the PACTEL facility. Quick look report. VTT Energy, TEKOJA 5/96.

Riikonen, V. 1996a. RELAP5/Mod3.2 analyses of the steam generator multiple tube rupture experiments (PSL02 and PSL04) on the PACTEL facility. VTT Energy, TEKOJA 2/96.

Riikonen, V. 1996b. Steam generator multiple tube rupture experiments on the PACTEL facility with the new steam generators. VTT Energy, TEKOJA 8/96, 20.12. 1996.

Riikonen, V., Puustinen, M. & Kouhia, J. 1995 RELAP5/MOD3.1 & APROS 3.0 analysis of SBLOCA in scaled VVER-440 geometry. Proceedings of the ANS winter meeting, San Francisco.

Semken, S. & Tuunanen, J. 1996. Experimental data report, PACTEL natural circulation experiments in low primary pressure. VTT Energy, TEKOJA 4/95.

Semken, S. 1996a. Quick look report. Pressure drop measurements of the passive safety injection system lines of PACTEL. VTT Energy, PAHKO 6/96.

Semken, S. 1996b. Quick look report. Pressure drop measurements of the passive safety injection system lines of PACTEL. Lines of PACTEL for GDE-31 through GDE-35. VTT Energy, PAHKO 8/96.

Tuunanen, J., Lillington, J., D’Auria, F. & Kalli, H. 1996. Review of experimental work and code application on the simulation of passive safety injection systems. VTT Energy, PAHKO 3/96.

Tuunanen, J. 1996a. Quick look report. Passive safety injection experiments GDE-21, GDE-22, GDE-23, GDE-24 and GDE-25. VTT Energy, PAHKO 2/96.

Tuunanen, J. 1996b. Experimental data report. Passive safety injection experiments GDE-21, GDE-22, GDE-23, GDE-24 and GDE-25. VTT Energy, PAHKO 4/96.

Tuunanen, J. 1996c. Minutes of the meeting of the planning group for small break leak tests (in Finnish). Helsinki, 20.6.1996.

Tuunanen, J. 1995. Assessment of RELAP5/MOD3.1 against PACTEL passive safety injection test GDE-11.VTT Energy, TEKOJA 12/95.

Tuunanen, J., Munther, R. & Vihavainen, J. 1996. PACTEL experiments for investigation of passive safety injection systems of advanced light water reactors. The 4th International Conference on Nuclear Engineering (ICONE-4). New Orleans, Louisiana, USA.

Tuunanen, J. & Vihavainen, J. 1996. Test specification report. Passive safety injection experiments GDE-31, GDE-32, GDE-33, GDE-34 and GDE-35. VTT Energy, PAHKO 7/96.

Vihavainen, J. 1996. Analyses of passive safety injection experiment GDE-24 with APROS 4.02 code. VTT Energy, PAHKO 9/96.

3.4 Severe Accident Management

The objective of reactor safety is to confine the radioactive substances generated in the reactor under all conditions. To achieve this, the defence in depth concept has been adopted in nuclear plant design and operation. Defence in depth implies multiple independent barriers. The reactor design recognises three main barriers: the cladding of the fuel rods, the primary system and the containment. Integrity of the barriers is secured by plant safety systems, designed to fulfill three main safety functions: to shut down the reactor, to remove the decay heat and to maintain a leaktight containment. A high reliability of the safety systems is achieved by utilizing redundant and diversified systems.

The hypothetical accidents used for safety system design are called *design basis accidents*. A plant should survive the design basis accidents without major release of radioactive substances, even if some safety systems fail to perform their functions. If the safety systems can not fulfill the first two safety functions the accident ultimately results in reactor core damage, a situation called a *severe accident*. A severe accident can only occur if a combination of unfavourable conditions exists. Severe accidents are extremely rare, for a modern LWR the estimated core damage probability is typically 10^{-5} per reactor year. In spite of the low probability, severe accidents have been extensively studied. This is because they are the only type of nuclear reactor accidents in which a substantial amount of radioactive fission products could escape from the containment.

Accident Management (AM) and *Severe Accident Management (SAM)* consist of several tasks. Introducing a SAM policy is a task for the safety authorities, and the utilities operating the nuclear power plants are responsible for the SAM implementation. The role of a research organisation is to provide tools and expertise for SAM strategy development, and to assist in selected tasks of SAM implementation.

The publicly funded Severe Accident Management project (VAHTI) 1994 - 96 was established to assist the current SAM programmes of the Finnish nuclear power plants. The chapter mainly describes the achievements of the VAHTI project. The activities of the VAHTI project are continued in the current Reactor Accidents Phenomena and Modelling (ROIMA) project, planned for the years 1997-99. The background of the VAHTI project were the thermal hydraulic and severe accident studies performed within the former Severe Accident (VARA II) and Accident Management (HOHTI) projects summarized in (Mattila & Vanttola 1995). The objective of the prior projects was to study the general progression of accidents in sequences relevant for the Finnish nuclear power plants.

The characteristic accident sequences having major contribution to core damage risk in Finnish nuclear power plants were considered well known when the VAHTI project was planned. The background were the comprehensive PSA level 1 studies available and the large number of plant specific accident progression calculations. The utilities operating the nuclear power plants had already implemented plant backfits and operating improvements to cope with severe accidents. The objective of the VAHTI project was therefore set to investigate in more detail some selected accident management questions. The results were needed to validate the SAM plans, to assist in their implementation and to reduce uncertainties of the severe accident consequence estimates.

The VAHTI project was divided into five main work packages:
- thermal hydraulic validation of the APROS code
- core melt progression within the BWR pressure vessel
- failure mode of the BWR pressure vessel
- aerosol behaviour experiments
- development of a computerized severe accident training tool.

3.4.1 Thermal hydraulic accident analyses

Thermal hydraulic phenomena play a central role in the progression of transients and accidents into core damage. They are also essential when assessing the plant behaviour in the severe accident phase. The thermal hydraulic computer codes are necessary tools in planning of efficient AM strategies and developing emergency operating procedures. To ensure correct understanding, the codes must be validated and the code users must be trained by calculating relevant test cases. Own code development creates deeper know-how and enables to focus in questions specific for the Finnish nuclear power plants.

RELAP5 has been the main analysis tool for design basis accidents since 80’s. The code has a full non-equilibrium six equation two-fluid model, which is solved by a partially implicit numerical scheme. The code is suitable for analysis of the full range of loss-of-coolant accidents and for certain operational transients. VTT and the other Finnish RELAP users have performed RELAP5 validation in the framework of the USNRC CAMP-programme. The main emphasis has been put to VVER-specific features, analyses supporting thermal hydraulic tests with the PACTEL-facility, and testing of the code capabilities in new operating conditions.

The Advanced Process Simulation Environment, APROS, has been jointly developed by VTT and Imatran Voima Oy (IVO). APROS provides tools, solution algorithms and model libraries for simulation of power plant processes, including the process automation and electrical systems. Thermal hydraulic processes can be alternatively described with homogeneous (3-equation), five, or six equation formulations. APROS allows the user to combine systems described with different level formulations into one model. One of the main tasks of the VAHTI-project has been thermal hydraulic testing of the APROS models. APROS based nuclear power plant analysers of the Loviisa and Olkiluoto plants have also been improved as a result of the project. The basic APROS development has not been a part of the VAHTI project, but has been conducted by a separate programme financed by VTT and IVO.

A set of tests of the interfacial friction models have been conducted by comparing the APROS models with tests on experimental facilities.

The IVO full-scale loop seal experiment (Tuomisto & Kajanto 1988) was calculated with the 5-equation model of the code version APROS 2.8 and with the 6-equation models of APROS 2.8 and 2.9 (Karppinen 1996b). The experiment was chosen as a high priority case, because correct modelling of a loop seal is essential for simulating the small break accidents in the VVER-440s. In case of a stratified flow, the onset of the slug flow occurs when water hits the top of the loop seal tube. The calculated water inclination was too steep with the APROS 5- and 6-equation models indicating that the interfacial friction was too high. Steady horizontal stratified flow caused numerical oscillations when using the 5-equation model. The cause was attributed to the calculation of the void profile of the drift-flux model.

The residual water level at the end of the experiment is a key parameter to test the loop seal modelling. The 5-equation model underpredicted the residual water level in all cases calculated. Water level calculation was improved in the 6-equation model as a result of the loop seal calculations.The final model predicted the residual water well for the whole range of gas velocities as shown in Fig. 47.

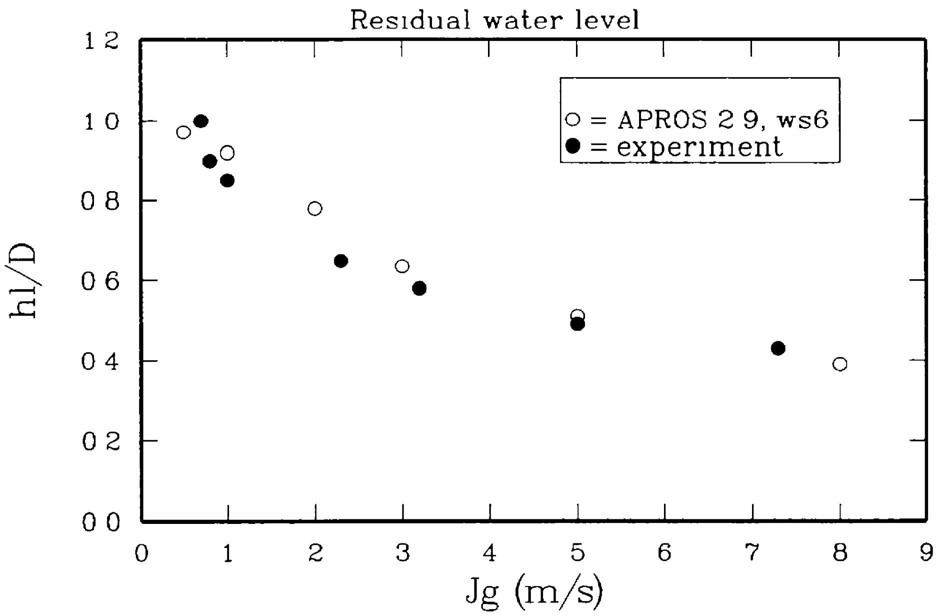


Figure 47. Residual water level in the loop seal experiment. APROS version 2.9 using the 6-equation model.

The LOTUS air/water annular flow experiments (Owen et al. 1985) were another source for interfacial friction model testing. The LOTUS test rig consists of a long (23 m) vertical tube to get fully developed flow conditions. The internal diameter of the tube is 31.9 mm. Water was fed on the wall of the test section with constant rate and the resulting film flow and pressure drop was measured with different air fluxes.

Two pressure loss and water entrainment tests with differing pressures were calculated with the APROS 2.11 code version assuming air and water (Karppinen 1996c). The steam tables used with the version 2.11 were not accurate at the low pressures used in the tests, and the calculation temperature had to be artificially increased from the actual 20 ^{o}C to 50 ^{o}C to get higher steam partial pressure. Generally, the results are in good agreement with the experimental data.

A simple numerical test was created to evaluate phase separation and interfacial friction model of the recent APROS version 4.04. The thought test consisted of a vertical pipe 0.9 m long and 1 m diameter supplied by gas and water from the lower end. The purpose of the calculations was to document the response of the APROS interfacial friction models in a wide range on conditions. The tests were run with the 6-equation model at three pressure levels: 0.1, 1 and 10 MPa, and three inlet velocities: 0.5, 2 and 10 m/s (Karppinen 1996a). The equivalent diameter and the flow geometry were also varied. A total of 36 test runs were performed. No discontinuities of the interfacial friction were found. Gas flow oscillations were observed in some cases with stagnant water. A physical explanation for the oscillations has been found (Karppinen 1996a). They may cause problems in some types of simulations with low pressures and non condensible gases. A most probable effect will be increase of the computing time.

OECD/CSNI international standard problem ISP-38 was an open exercise based on test 6.9c performed with the BETHSY facility. The test belongs to a program, in which the objective is to understand the PWR response during a loss of the Residual Heat Removal System (RHRS) at a mid loop condition. The test was begun with the primary circuit liquid level at the axis of the hot legs. The accident scenario simulated a failure of RHRS while the pressurizer manway and the outlet plenum manway of the steam generator on one loop were open. The specific initial conditions included atmospheric pressure without condensable gases and 100 °C coolant temperature.

The transient had four phases. The two phase level in the pressure vessel is situated close to the axis of hot legs in the initial phase. The RHRS circulation was interrupted in the second phase and core boiling started. The two phase level falls below the elevation of the hot leg nozzles during the third phase, and core heat-up is initiated after the mixture level has reached the top of core. The fourth phase begins when a gravity feed injection is started after the core temperature reached a specified value. The injection of the test was sufficient to rapidly reflood the core and refill the whole pressure vessel up to the mid loop condition.

The APROS 5- and 6-equation models and the RELAP5/MOD3.2 code were applied in parallel to calculate the test case. The RELAP5 results are discussed in context of the RELAP5 validation. The atmospheric boundary conditions of the test required revision of some APROS models. The main causes for deviations from measurements were phase separation in the core, horizontal stratification in the hot leg and the counter current flow limitation in the pressurizer surge line and steam generator inlet. Predictions of the 5- and 6-equation models differed considerably from each other. The 5-equation model calculated high upper plenum pressure and high fluid discharge from the system early in the transient. This caused early cladding heatup and triggering of the safety injection. System pressures were overpredicted in all phases of the transient. The upper plenum pressure calculated by the 6-equation model was nearly constant throughout the transient. Consequently, the measured pressures were underestimated early in the transient and overestimated later on. The calculated mass inventory of the system follows closely the measured behaviour. Due to the good agreement in mass balance the core heatup and triggering of the safety injection were well predicted.

Validation of the RELAP5 code

Validation of the RELAP5 code has been done within the framework of the USNRC Code Assessment and Maintenance Project (CAMP). VTT has performed analyses of the PACTEL Passive Safety Injection Test GDE-11 and the OECD/CSNI International Standard Problem ISP-38. IVO and Lappeenranta University of Technology have analyzed IVO CCFL test with full-scale fuel bundle structures and IAEAs SPE-4.

The PACTEL test facility is described in section 3.3. The passive safety injection test series GDE-11 ... GDE-14 studied the Core Make-up Tank (CMT) behaviour. Test GDE-11 simulated a 2 % cold leg break with 1.8 % scaled decay power and ECC water supplied only from the passive safety injection. The passive injection system consisted of the CMT, connected to the facility with an injection line and two pressure balancing lines.

Rapid steam condensation in CMT was seen in the test. A thin layer saturated water formed in the top part of the tank above a bulk of cold water. Break up of the top layer caused condensation, resulting in a fast pressure drop in CMT and stagnation of the ECC flow (Tuunanen & Kouhia 1995). General view of the PACTEL facility has been presented in Fig. 38 in chapter 3.3.

The case was calculated with the RELAP5/MOD3.1 code (RELAP Development Team 1995). The main discrepancy was that the code was not capable to simulate the rapid condensation in CMT, because the observed temperature stratification could not be correctly modelled. The results were very sensitive to the nodalization of the CMT due to the importance of conduction between stratified layers of water.

The standard thermal hydraulic characteristics of the test were predicted quite well. An example is the decrease of the primary pressure after the break opening. Water levels both in the upper plenum and in the pressurizer were overestimated. The calculation predicted a period of single phase natural circulation through the cold leg PBL, which was not observed in the test.

The OECD/CSNI ISP-38 was also calculated with the RELAP5/MOD 3.2 code. The test case and APROS results have been described in connection with the APROS validation. All major events of the test phases were predicted rather well by the RELAP5/MOD3.2 code, demonstrating that the code is suitable for use under atmospheric pressure conditions. RELAP5 calculated too large entrainment in the surge line, restricting steam flow through the pressurizer. This resulted in too high primary system pressure causing a loop seal to clear. None of the loop seals cleared in the test. Triggering of the gravity feed safety injection was occurring late in calculation. The collapsed level at this time was significantly under predicted. Previously, RELAP5 calculations at low pressure have had problems with mass error. This calculation with the most recent development version (MOD3.2.1.2) showed however very low mass error, about 0.4 % of the primary mass inventory.

3.4.2 BWR core degradation

Consequences resulting from operator actions after the reactor core has been partially damaged have been systematically assessed in a series of studies for the Nordic BWRs. The studies have been conducted as a part of RAK-2 project within Nordic Nuclear Safety Programme (NKS) 1994 - 1997. The work has been divided into three subtasks: reflooding, recriticality and late phase melt progression, and has been carried out by partners from SKI, Studsvik Eco & Safety AB, Vattenfall Energisystem AB, Risö National Laboratory, Teollisuuden Voima Oy and VTT Energy.

The objectives have been to evaluate when and how the reactor core is still coolable if safety systems become available during the accident, and what are the probable consequences of water cooling. The emergency core cooling water in the Nordic BWRs does not contain boron, which raised the question of BWR recriticality. The next step is to investigate if the core debris reaches a coolable configuration in the reactor pressure vessel (RPV) lower head, and the possible failure mode of the RPV lower head if debris cooling is insufficient.

Reflooding of a degraded core

Reflooding of an overheated BWR core was investigated by calculating accident scenarios for ABB Atom BWRs with three different computer codes. The results have been presented in detail in (Lindholm et al. 1995a and Lindholm et al. 1995b). The realism of the calculated results was evaluated by comparison with experimental observations. The calculations were carried out for the Finnish TVO and Swedish Forsmark power plants, which resemble each other and use same type of fuel and control rods, but have different thermal powers and different alignments of safety injection systems. The codes applied were risk analysis codes MAAP4 and MELCOR 1.8.3, and the detailed best-estimate code SCDAP/RELAP5.

The base scenario was station blackout with two variations:

1) successful depressurization of the reactor coolant system (RCS) and
2) failure of the automatic depressrization system (ADS).

The former case results in low RCS pressure during core degradation, in the latter case RCS pressure remains high until the possible failure of pressure vessel. A variety of times was assumed for restoration of power and subsequent start of coolant injection. The criterion used was the maximum core temperature, varied from 1400 K to 2000 K. The important phenomena related to rewetting of a degraded core include rapidly increased hydrogen generation and the recriticality considerations. The studies focused on the following issues:

- how late can the reflooding start to maintain the melt within the core boundary
- what is the effect of core top spray cooling compared to filling up the core from below
- how much hydrogen and heat is released during reflooding
- what is the time window between control rod relocations and fuel melting.

All the codes predicted a different core end state after reflooding: MAAP4 predicted formation of melt pool in the core, MELCOR resulted in formation of rubble bed and SCDAP/RELAP5 predicted material melting and/or fuel fragmentation due to mechanical stresses caused by temperature difference of coolant and hot fuel rods. Rapid core cooling was obtained in cases when reflooding was started at maximum core temperature of < 1600 K even with half of the total capacity of the high pressure injection system. The core was still coolable if the maximum cladding temperature at the beginning of reflooding was < 1800 K, but core cooling was slower. The core end states calculated with MELCOR and MAAP4 are compared in Fig. 48.

High pressure cases resulted in large local temperature differences due to slow boiloff of coolant and respective efficient oxidation in the uncovered parts of the core. In all reflooding cases the emergency coolant injection increased hydrogen production. Typically the oxidation heat exceeded the decay heat generation. If the core was rewetted from below with downcomer injection, the maximum core temperature had a sharp increase prior to start of quenching. This temperature peaking was caused by rapid cladding oxidation in the upper parts of the core.

All codes predicted some time window between melting of control rods and fuel. The lenght of the time window varied with codes, being 1 - 2 min with MELCOR and 3 - 15 min with MAAP4. All codes predicted that there is at least a metastable period, where part of the core (10 - 50 %) is without control material. During this period the fuel pellets may be in a rubble bed or in intact fuel geometry. Recriticality studies were judged to be most interesting in cases, where fuel rods are in intact geometry and the control rods have partially melted.

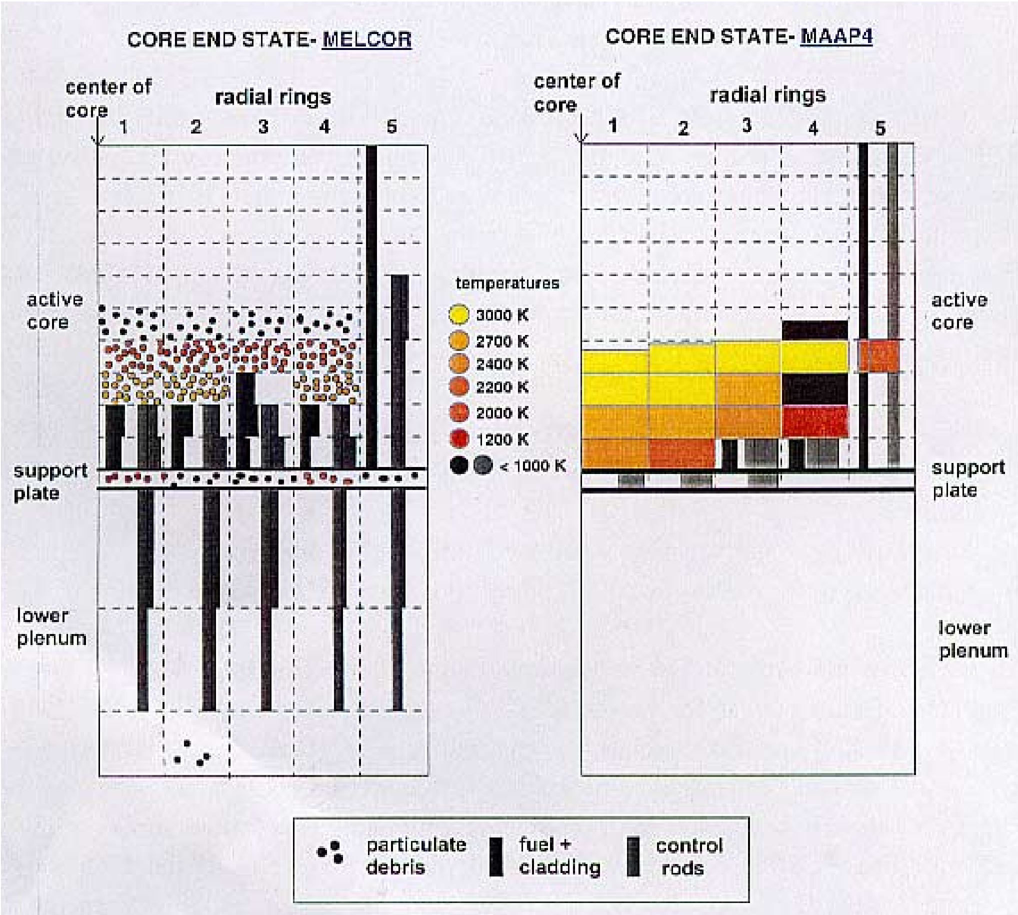


Figure 48. End state of the core in the TVO low pressure case with reflooding to downcomer. Reflooding was assumed to begin when maximum cladding temperature exceeded 1800 K. MAAP4 and MELCOR predictions (Lindholm et al. 1995a).

Recriticality in degraded core

Reactivity accidents of a degraded core have been addressed in some studies during the past decade with rather different conclusions (Okkonen et al. 1993, Scott et al. 1990). In the earlier work by Scott it was concluded that recriticality is possible but the power excursion would be moderate and not able to fragment the fuel. Furthermore, if the reactor would remain critical, the power would stabilize at or fluctuate around an elevated power level estimated to be about 10 - 20 % of full power. Okkonen, on the other hand proposes that a degraded BWR core will reach recriticality if at least one meter height of the fuel is intact and the coolant void fraction in the core is below 0.6. Two-dimensional four-bundle studies by Mosteller et al. 1995 suggest that retention of 10 - 20 % of control poison may prevent recriticality. However, three-dimensional effects may affect the result, since for example all the present severe accident models predict non-uniform melting of the control rods. Shamoun et al. 1994 presents that the loss of 95 % of control rods may lead to recriticality during reflooding phase. According to Shamoun boron concentration of 1200 ppm in coolant would prevent recriticality. Earlier studies based on simplified calculations (Anttila 1991) suggest that 850 - 1000 ppm boron in the coolant would be sufficient to prevent recriticality in intact fuel geometry of Olkiluoto 1/2.

Joint Nordic recriticality studies are currently under way. The first results have been reported in (Højerup et al. 1997). The Nordic studies will use three different reactor physics tools. A simple model, called RECRIT, has been created by Risö National Laboratory and VTT Energy. It was created by coupling of the neutronics written for the original RECRIT code with the thermal hydraulic models developed at VTT Energy. The upgraded thermohydraulic model accounts for the quench front movement and the heat transfer below and above the quench front as well as the heat generation due to cladding oxidation.

As a second approach, APROS simulation code will be applied to study recriticality at VTT. A three-dimensional code model for the Olkiluoto reactor has been prepared describing 500 fuel bundles in 250 core channels. The initial core damage states for the recriticality calculations are taken from the MELCOR and MAAP4 calculations.

The third method to study recriticality will apply the SIMULATE-3K code for neutronics. SIMULATE-3K is a transient reactor physics model with more detailed neutronics but simpler thermohydraulics than e.g. APROS. The SIMULATE-3K studies will be performed by Studsvik Eco & Safety AB with Forsmark 3 plant data. A tentative calculation has been performed with a simple core model having 4 x 4 BWR assemblies. The case studied started with total station blackout followed by scram and stopping of core flow. The water boil-off and core heat up started, and when the core temperatures had increased to 1500 K, half of the number of control rods were withdrawn and reflooding was started. The tentative results showed that the Doppler feedback limited the power pulse to less than one full-power second, although high power peaks may occur. After that peak, fission power stabilized at an elevated level, the magnitude of which increased with reflooding flow rate. In case of water injection of 625 kg/s corresponding to 5 % nominal feedwater flow rate, the power stabilized at about 35 % of full power.

An application case studied with the RECRIT code was a BWR accident initiated with total loss of electricity. This trips the reactor and the turbine and switches off the circulation pumps. A natural circulation pattern is soon established. Energy is removed from the core by slow boiling through the steam relief valves. System depressurization is initiated by opening of the ADS dump system. The core is uncovered during the ADS blowdown and the core heatup commences.

Several coolant injection rates were assumed varying from 22.5 kg/s representing capacity of a single high pressure injection pump to 540 kg/s, the capacity of all four low pressure injection pumps. When reflooding water fills the core a significant fraction of the incoming coolant is entrained above the quenching front, forming a droplet-dispersed flow. The movement of the quenching front defines the point, where a neutron flux can peak. Criticality may be expected when the quenching front reaches a core zone without control materials.

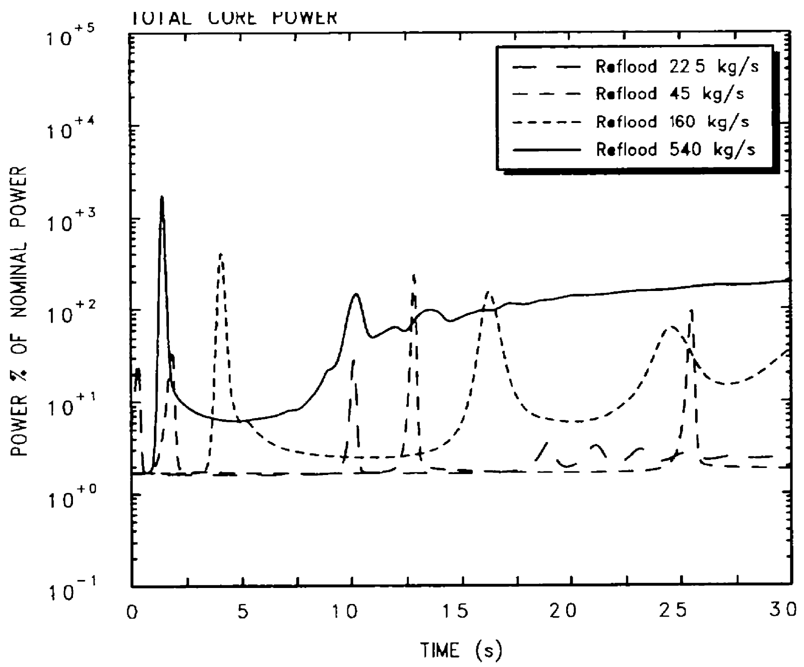


Figure 49. Total core power generation calculated with the RECRIT code (Højerup et al. 1997).

The calculations resulted in reasonable trends in power generation during recriticality. The reactor power peaks rapidly after reaching criticality and then stabilises or oscillates at an elevated power level, as shown in Fig. 49. The coolant void fraction and the fuel temperature are the principal factors affecting criticality, hence variations on coolant flow rate were carried out. Results of the flow rate variations have been collected in Table 6.

Table 6. Power peaks due to reactor recriticality calculated with the RECRIT code (Højerup et al. 1997).

Reflood flow rate (kg/s)	The maximum power peak *)	The other peaks *)	Stabilised power *)
22.5	1st peak, 30 %	2nd peak, 30 %	-
45	2nd peak, 220 %	1st peak, 30 % 3rd peak, 100 %	-
160	1st peak, 400 %	2nd peak, 100 % 3rd peak, 60 %	25 %
540	1st peak, 1700 %	2nd peak, 200 %	200 %

*) Power given relative to the core nominal power

Core melt behaviour in the lower head

If the overheated core cannot be quenched in the core region, the accumulated melt fails the core support plate and slumps into the reactor pressure vessel (RPV) lower head water pool. The BWR lower head houses a number of instrument tubes and control rod guide tubes with penetrations in the lower head bottom. The penetrations have been considered the most vulnerable locations concerning lower head integrity. The key debris parameters in the evaluation of lower head failure mode are morphology, chemical composition and temperature. An investigation of debris pool formation into the lower plenum has been initiated. The studies will be performed with parallel code calculations (MELCOR and MAAP4). A number of MELCOR calculations have already been performed utilising the optional detailed bottom head (BH) model developed at Oak Ridge National Laboratory.

MELCOR code typically estimates that debris is fragmented and quenched while passing through the lower head water pool after core support plate failure. This is followed by slow boil-off of lower head water. The initial debris bed porosity and debris particle diameter are user defined parameters. The BH model recognizes three debris layers. The lowest level has usually high metal contents (2/3 of its mass). The second layer consists mostly of oxides (UO_2). The third layer comprises initially mostly oxides, but after heatup and melting of core shroud, the metals fraction increases. The porosity of debris decreases when the different material components melt and flow downwards in the rubble bed.

Both low and high pressure scenarios for the Olkiluoto BWRs have been investigated (Lindholm 1996). In general, reflooding after lower head dryout speeded up the event of lower head failure. This was caused by excessive oxidation heat in the debris bed, which overrides the cooling effect of water and steam. The fraction of oxidized zirconium in the lower head was high in all reflooding cases: 47 … 81 % with the highest oxidation fractions being achieved in the high pressure cases. In a typical case, the lower head integrity was lost by failure of instrument tubes at several locations 60 cm above the bottom of the vessel. The interval between lower head dryout and failure of the instrument tubes varied from 13 min to 70 min with differerent debris particle sizes and initial porosities assumed. If the instrument tubes were not allowed to fail, the lower head failure occured via pressure vessel creep rupture significantly later (about 5 h after lower head dryout). The tentative analyses showed that reflooding after lower head dryout will not be able to prevent instrument tube failure.

3.4.3 Development of the PASULA code

The PASULA code system has been developed for the pressure vessel lower head analyses in severe accidents by VTT Energy. A schematic of the problem to be solved is depicted in Fig. 50.

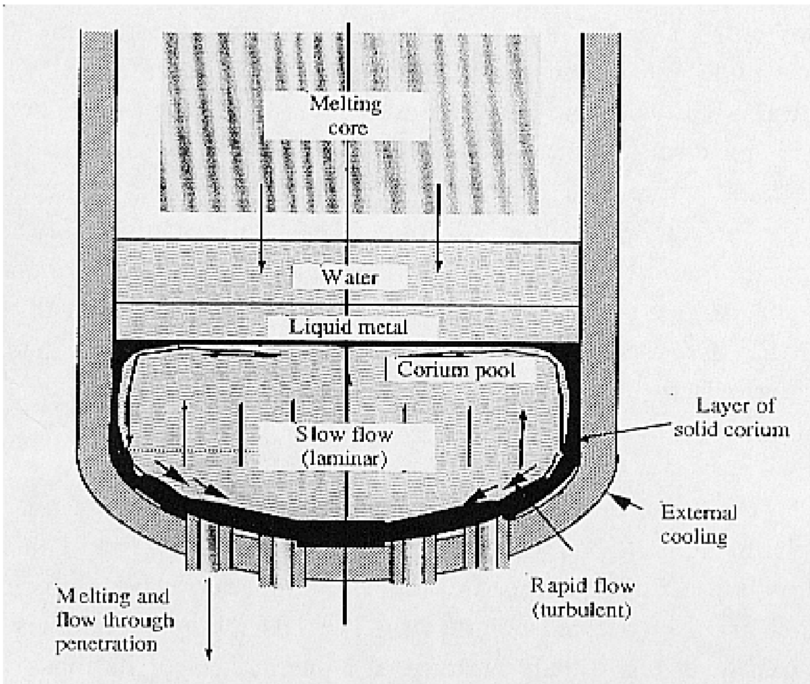


Figure 50. A schematic of the applications planned for the PASULA code system.

The PASULA system consists of a number of codes developed for specific tasks. Heat conduction and convection analysis can be performed for two- and three-dimensional cases. The codes in question, PASULA2D and PASULA3D, are based on finite difference and control volume method. The codes are nonlinear and take into account phase changes. The structure to be calculated may contain internal gaps, over which various heat transfer modes can be taken into account. Heating or cooling effect of fluid flow in narrow gaps can be analyzed.

Analysis of the solid continuum is performed by 2D and 3D finite element codes EPFM2D and EPFM3D. The codes are capable for calculation of elastic and plastic deformations due to combined mechanical and thermal stresses. An arbitrary geometry can be modelled with the three-dimensional system by applying 20-point isoparametric elements. The physical modelling is based on von Mises yield condition and the associative flow rule. The axial stress-yield function can be given as multi-linear representation or as a Ramberg-Osgood fit. The mathematical solver used in the codes has capabilities similar to commercial FEM codes.The codes are also suited for fracture mechanical analysis, like calculating the J-integral of semielliptical surface cracks in the reactor pressure vessel in vessel cooldown cases. Analysis of shallow cracks can also be performed. These additional capabilities would be required in a comprehensive assessment of the reactor pressure vessel external cooling, where the large thermal gradients are formed across the pressure vessel wall.

The PASULA code has been applied to calculate tests conducted at the Swiss Corium Reactor Vessel Interaction Studies (CORVIS). It has also been utilized in detailed studies of the ABB Atom BWR pressure vessel bottom head penetrations in core melt accidents. The objective was to compare the control rod and instrument nozzle behaviour and to evaluate their potential for initiating a vessel failure.

The pressure vessel bottom includes 121 control rod nozzles and 50 nozzles for neutron monitors and other instrumentation. Schematics of the control rod and instrumentation nozzle areas are presented in Fig. 51 and 52.

The thermal sleeves of the control rod nozzles are attached at their lower ends to a common steel plate, which is not supported by other structures. The weight of the control rod system is resting on the weldings, located within the reactor pressure vessel. The length of the instrument tubes below the pressure vessel bottom is about 5.5 meters. An individual instrument nozzle is restrained only by the welding located within the pressure vessel.

The instrument tubes differ somewhat from each other. The source range monitor and intermediate range monitor detector housing tubes probably contain steam at primary system pressure in a core melting accident. The local power range monitor detector housing tubes contain water, four copper wires and in the middle a Traversing Incore

Probe (TIP) tube. In a core melt accident the TIP tube is expected to melt at its upper part. It was assumed to be open to reactor pressure, and filled with steam escaping from the pressure vessel. The amount of steam flow is small, because the inner diameter of TIP tube is only about 7 mm.

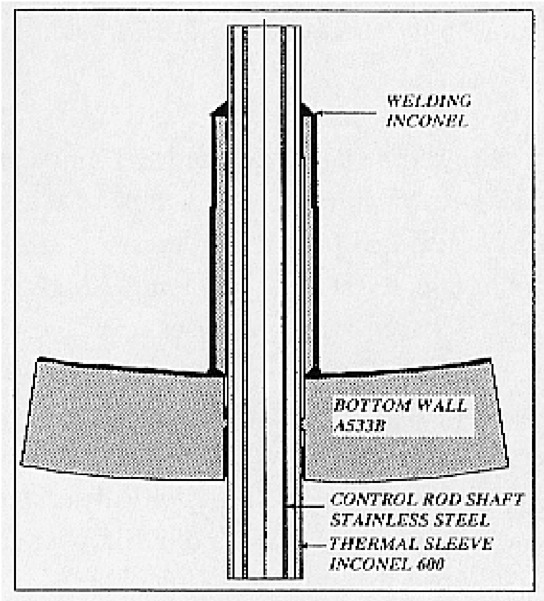


Figure 51. A schematic of a control rod penetration.

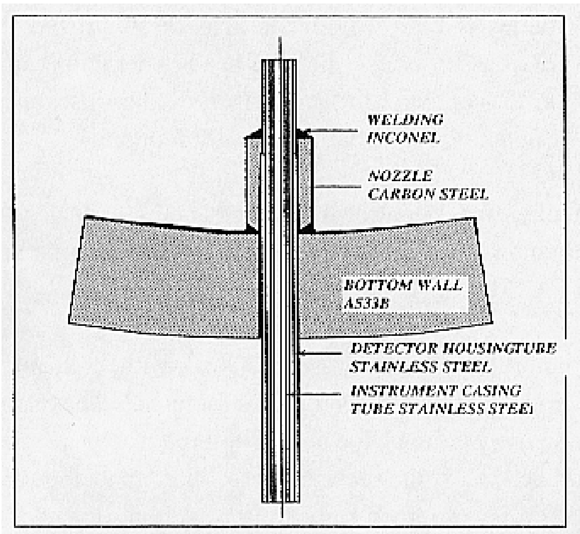


Figure 52. A schematic of an instrument tube penetration.

The analyses were axisymmetric. No natural circulation flow of corium nor convection was taken into account. The nozzle areas were idealized to be formed of rectangular axisymmetric blocks. Triangular shape of the weld was idealized as a rectangle having same area as the triangular cross section.

Control rod nozzles

The most critical part in the initial phase of a control rod damage is the thermal sleeve above the carbon steel nozzle. The control rods normally contain water in the annular space between the control rod shaft and the thermal sleeve, and presence of water was also assumed in the calculations. If surrounded by melt, the sleeve wall temperatures achieve a steady state after about 100 seconds. At this time, the calculated inner surface temperature was 1020 K and the outer surface 1520 K. The sleeve wall has at this time still enough strength to carry the assumed loads due to gravity and the 6 bar reactor pressure assumed.

The water may boil away from the annular gap, and bursting of the thermal sleeve can hence not be excluded. If it occurs, the hot melt will flow into the control rod annular space. The calculations show that this will be followed by a rapid loss of strength of the control rod shaft due to the small wall thickness and the stainless steel shaft material. Failure of the shaft must be assumed only within some tens of seconds.

The corium flowing downwards cools and shrinks. At the same time, the tube will be heated and expands. A possibility exists, that contact between corium and the thermal sleeve will be lost. Analyses with a gap between crust and the thermal sleeve were therefore performed. It was found out, however, that a gap between corium and the sleeve did not essentially change the results, because most of thermal energy from the crust to the thermal sleeve tube is always transported by radiation.

The calculations indicated that the thermal sleeve would lose strength at the level of the vessel insulation layer after about 500 seconds in case of a gap and after about 200 seconds in case of no gap. The corium within the tube will be cooled below 2000 K at this time, be solid and have some strength. The thermal sleeve and control rod shaft tubes are all fixed together at their lower ends and the intact tubes will support the broken ones. In absence of horizontal forces a large corium leakage through the control rod nozzle is hence not likely.

Calculations for the instrument nozzle

Initially, the corium will attack the detector housing tube. The amount of corium on the vessel bottom was found to have a negligible influence, because of the small heat capacity of the tube. The calculations, demonstrated in Fig. 53, showed that the tube wall temperature reaches its maximum in about 60 seconds, from which moment on wall temperatures stay nearly stationary.

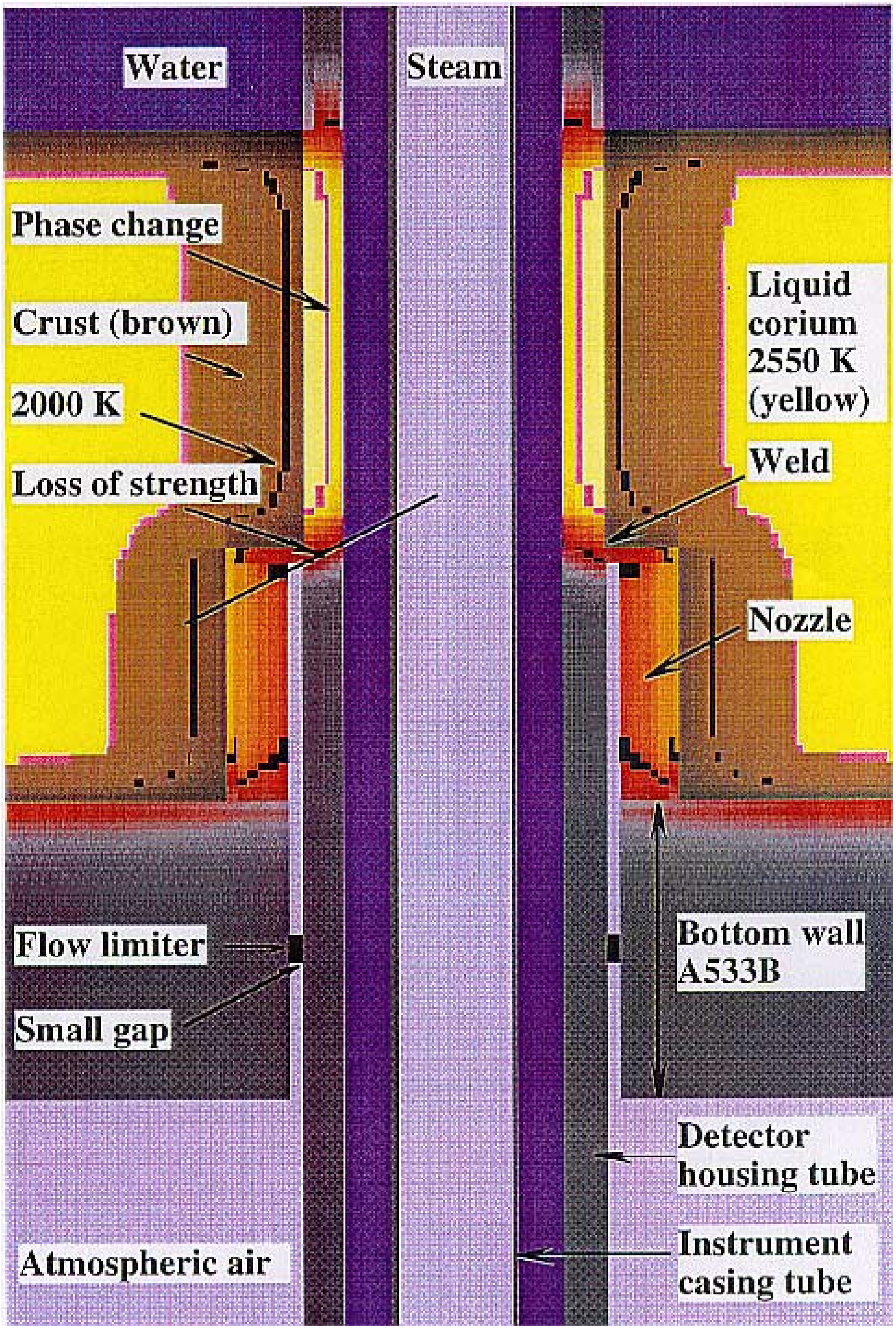


Figure 53. Thermal analysis of an instrument tube penetration. Situation 40 s after corium ejection into the reactor pressure vessel lower head.

Most of the wall is in liquid phase at this time. A good contact exists between the tube and corium due to the tube thermal expansion. The most critical point of the housing tube is located just below the welding. There is no pressure difference over the tube wall. However, the margin to liquid temperature of stainless steel is so small that even a small load can break the thin solid layer. It was hence assumed that the housing tube fails at this point, after which the tube will fall out. The viscosity of liquid corium is low and corium is able to rapidly fill the annular space between tubes. In the next phase, the instrument casing tube heats up. Because of the small wall thickness this occurs very quickly, in about 20 seconds. Liquid corium will then fill the casing tube too, and flow downwards. Even if this flow could be initially contained within the tube, thermal analysis showed that the strength of the tube will be lost after some tens of seconds at the level of the RPV insulation.

The PASULA calculations have shown that the construction of a nozzle has an essential effect on the RPV failure mode. Main factors are the location of load carrying points, wall thickness and the material involved. It was concluded that a large corium leakage through the control rod penetrations is not expected for this type of BWR pressure vessel. If integrity of the control rod thermal sleeve is lost, corium flowing into the tube will be cooled and solidified prior of a tube failure on the external side of the RPV. The control rod tubes are supported at their lower ends by a common steel plate, due to which a broken tube will not easily fall down. The walls of an instrument nozzle tube fail quickly, within some tens of seconds. The corium flows then along the annular space between the detector casing tube and the instrument tube and within the instrument tube. The instrument tubes are not supported at their lower ends. The falling out of an instrument tube is much more probable than that of a control rod tube. The reactor pressure vessel failure is hence expected to be initiated at the instrument tubes.

It can also be deduced that external cooling of the pressure vessel could not prevent failure of the instrument tubes. Heating of the welding and the tubes happens quickly within the vessel and far from the external surface of the pressure vessel. The results were not sensitive to the amount of corium on the vessel bottom head. Failure of the instrument nozzles is basically a local phenomenon, because of the small heat capacity of the nozzles.

Creep modelling

A recent addition to the PASULA system is modelling of material creep. A detailed descprition of the model has been published (Ikonen 1996) including some test cases.The main assumptions used in the model development are:

- the basic equilibrium equations have been derived from the principle of virtual work
- the material properties are acquired from one-dimensional stress tests

- the three-dimensional stress state is reduced into one dimension using the isotropic von Mises model. This is used to determine the yield point and the creep rate from the empirical one-dimensional tests
- the plastic deformation and creep are assumed to occur in a direction perpendicular to the yield surface
- the deformation work per unit volume is assumed to be equal to that in an one-dimensional axial test
- as a final step, the calculated stress-strain relation will be retransformed into three dimensions.

The numerical stability of the solution method has been extensively tested. The code has also been applied to calculate creep of Zircaloy fuel rod claddings under reactor conditions.

Plans for the future development and applications include first validation of the creep analysis method for reactor pressure vessel type of stainless steels. The code testing will start with one-dimensional cases, to be extended later with three-dimensional pressure vessel model tests. The ultimate goal is to predict the failure mode of a reactor pressure vessel under all important severe accident scenarios.

3.4.4 Hygroscopic aerosol behaviour in LWR containment conditions: AHMED experiments

Aerosol behaviour in the containment is an important factor in determination of the possible release of radioactive materials into the environment following a severe accident, the so-called source term. If the containment fails or leaks, the source term is directly related to the concentration of radioactive aerosols in the containment atmosphere. Fig. 54 is a schematic of the different mechanisms affecting the aerosol concentration in the containment. The figure does not address any particular reactor type or accident scenario. The major uncertainties have been determination of aerosol sources, as well as modelling containment thermal hydraulics and aerosol behaviour.

Large discrepancies have been observed between the results from aerosol model calculations and measurements obtained from large scale integral tests (e.g. Wilson et al. 1988 and Haschke & Schönck 1988). Model validations with small scale experiments, where the boundary conditions are accurately controlled were considered necessary. The objective of the AHMED experiments was to understand the principal reasons for the observed large discrepancies between results from model calculations and measurements. The single compartment experiments in AHMED and PITEAS facilities (Mäkynen et al. 1994, Sabathier et al. 1994) have greatly reduced these uncertainties.

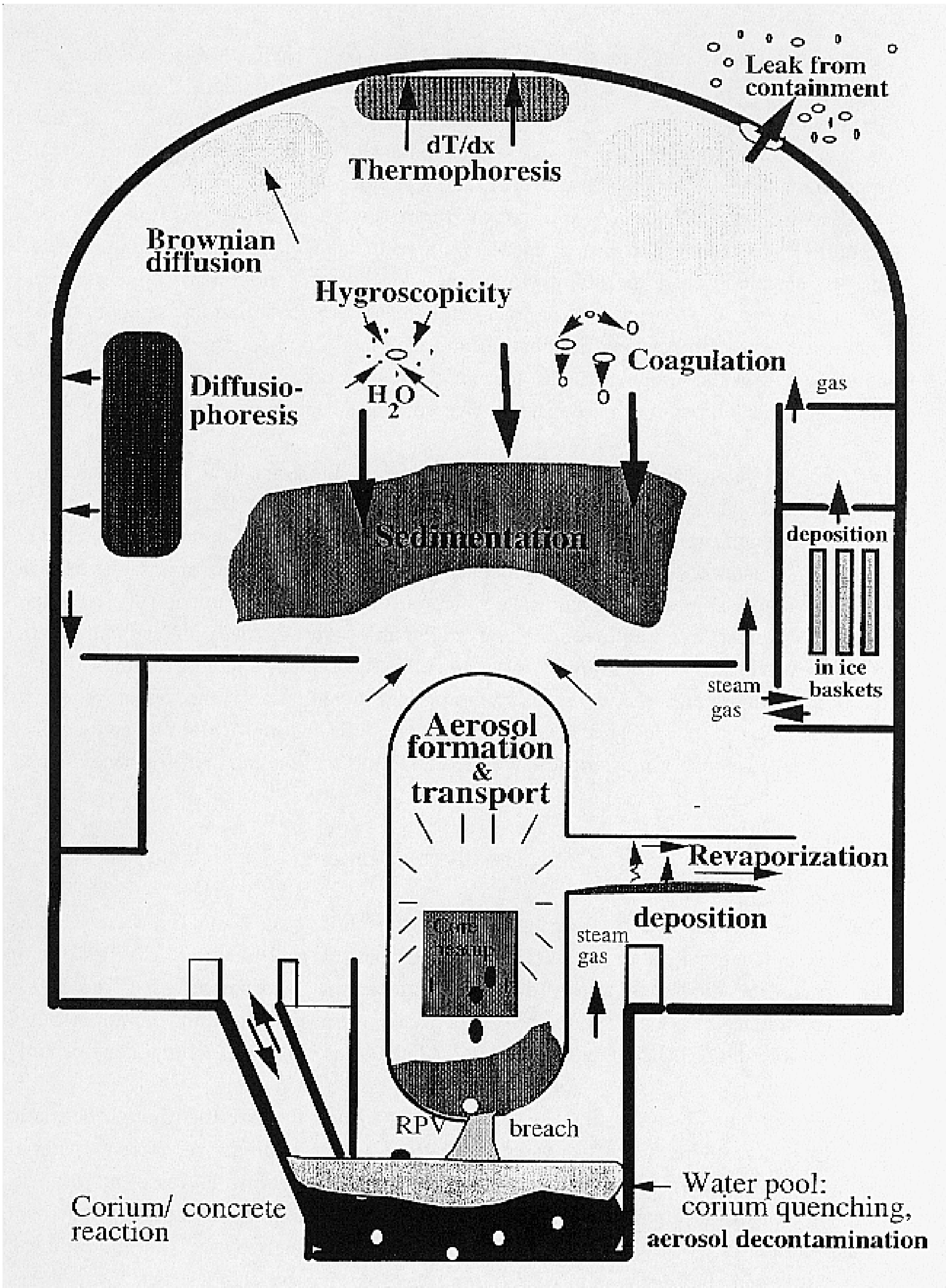


Fig. 54. Aerosol removal processes in a PWR containment

The Aerosol and Heat Transfer Measurement Device (AHMED) was constructed by VTT Aerosol Technology Group to study the effect of thermal-hydraulics on hygroscopic and inert aerosols in containment. The geometry of the AHMED test vessel is simple consisting of a 1.81 m^3 single compartment tank. The thermal-hydraulic conditions of the AHMED vessel were accurately controlled together with known steam and aerosol injection rates. At well known thermal hydraulic conditions it is possible to minimize the feedback from thermal-hydraulics to aerosols and validate the aerosol models separately from thermal hydraulic models. State-of-the-art aerosol measurement systems were used. Aerosol number and mass concentration was continuously measured. The particle size distribution and chemical composition were analyzed by impactors. Special emphasis was placed on temperature measurements, because a homogeneous temperature field inside the vessel was a necessity for the experiments.

Wet conditions prevail in LWR containments during most accident sequences. In this case steam condensation on aerosol particles, coupled with particle agglomeration, can provide a much more effective growth mechanism than dry agglomeration alone. The growth is enhanced, because the aerosol released from the core contains hygroscopic material (for example CsOH and CsI), which absorbs water even under 100 % relative humidity. Under favourable conditions particles may grow to sizes larger than 10 µm. These particles will settle rapidly, after which only a small fraction of micron-sized particles remain airborne. The ability to model quantitatively the effect of steam condensation on particles depends strongly on both thermal-hydraulic and aerosol modelling. It is also important to have the capability to calculate aerosol behaviour in multicompartment systems.

Conduct of the experiments

NaOH, CsOH, CsI and Ag aerosols were used. NaOH, CsOH and CsI were used as hygroscopic aerosol materials. Ag was used as inert aerosol material. The first set of experiments studied behaviour of different chemical species separately. In a second set of experiments, Ag and CsOH aerosol particles were generated and simultaneously injected into the AHMED facility to study the behaviour of the multicomponent aerosol.

The aerosol was always generated in the same way and the size distribution checked. Spherical aerosol particles were generated from water solutions of aerosol material using ultrasonic aerosol generators. After the aerosol injection the vessel atmosphere was mixed with a propeller having a very small settling area. After mixing the aerosol mass and number concentrations were monitored continuously over about 300 minutes. From this data the decay rate of different aerosols can be characterised by a decay constant.

It was verified that all vessel internal surface and gas temperatures were approximately equal. The maximum difference in gas temperatures was about 0.4 ^{O}C and stayed constant during the experiments. The relative humidities were nearly equal at different locations within the vessel.

The vessel gas temperature, internal and external surface temperatures were measured in several locations. The external temperatures as well as pressurized air and steam temperatures were also measured. The relative humidity was measured using three detectors. The vessel and input line pressures and steam and air flows were also monitored continuously. Input air flow was filtered and dried. The vessel surface temperature was controlled using computer controlled heating cables. The input gas mixture temperature was regulated using a heat exchanger.

Aerosol number and mass concentrations were continuously measured after drying the aerosol in a diluter and in a diffusion dryer. The particle size distribution and chemical composition was analyzed by four 11 stage low pressure impactors in the aerodynamic diameter range 0.03 - 15 µm. A schematic diagram of the aerosol measurement system and the thermal hydraulic measurement points is presented in Fig. 55.

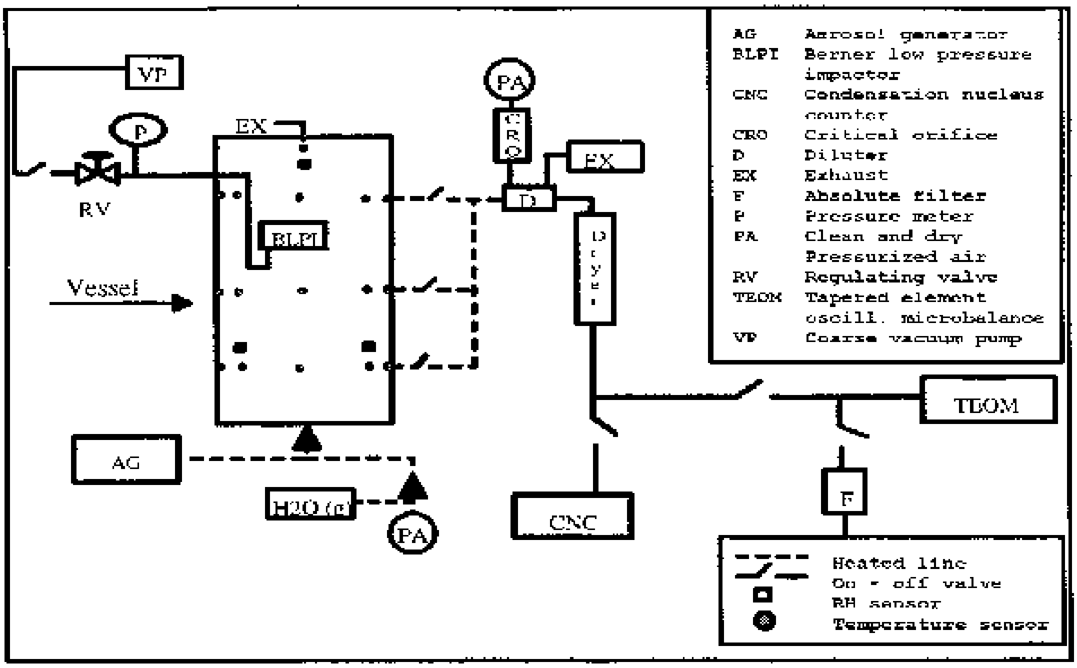


Figure 55. The main AHMED measurements.

Measured aerodynamic mass median diameters and geometrical standard deviations of the particle size distributions are presented in Table 7.

Table 7. Measured dry aerosol size distributions (based on impactor and APS measurements).

Aerosol material	AMMD [µm]	GSD
NaOH	2.4	1.64
CsI	2.3	1.7
CsOH	2.1	1.7
Ag	2.7	1.7

Mass and number concentrations measured from three different heights were equal within the limits of the measurement accuracy. Temperatures and relative humidities (RH) during the experiments and mass concentrations in the beginning of the experiments are presented in Table 8.

Table 8. The AHMED experiments.

Experiment:	Relative Humidity (%)	Temperature (C)	Initial mass concentration (mg/m^3)
NaOH 1	22	50	112
NaOH 2	34	20	200
NaOH 3	82	27	208
NaOH 4	91	17	638
NaOH 5	96	23	218
CsI 1	27	23	102
CsI 2	82	24	98
CsI 3	96.5	27	94
CsOH 1	7.3	51	89
CsOH 2	33	19	86
CsOH 3	39	19	73
CsOH 4	96	23	64
CsOH 5	97	28	94
Ag	54	22	79
Ag+CsOH 1	91.5	24	82
Ag+CsOH 2	91	25	112
Ag+CsOH 3	97	24	60

Experimental results

NaOH experiments

The NaOH experiments were carried out at atmospheric pressure and constant relative humidities (RH 22%, 34%, 82% 91% and 96%). The maximum temperature was 50^{o}C. The mass concentration varied at the beginning of the experiments. The number concentration was so small that sedimentation enhanced by hygroscopic particle growth was the only effective deposition mechanism.

Normalized NaOH mass concentration time behaviour at different relative humidities measured from the middle height are shown in Fig. 56.

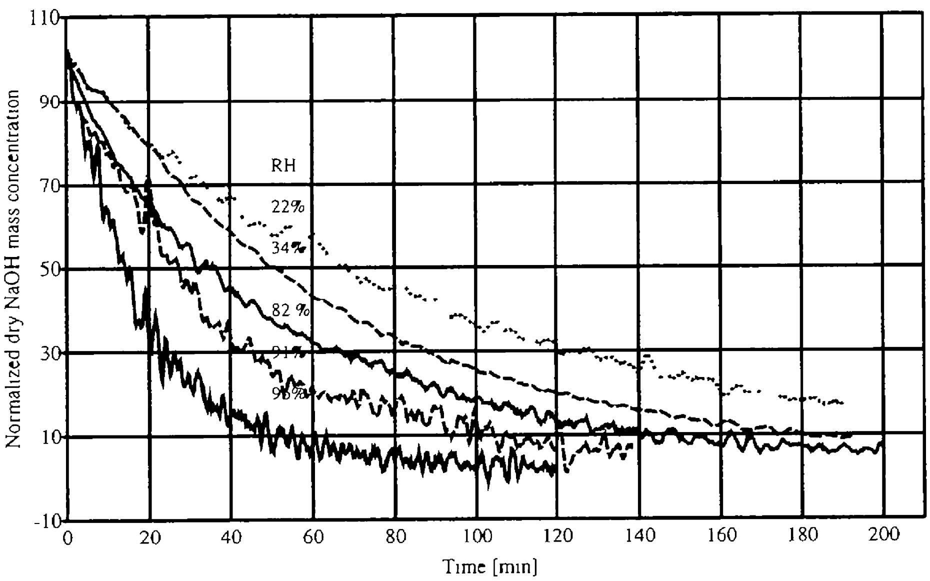


Figure 56. The normalized NaOH mass concentration as measured in the AHMED facility (Mäkynen & Jokiniemi 1995).

The ratio of the aerosol mass concentration half lives at low to high relative humidity experiments was about 4.

CsOH , CsI and Ag experiments

The CsOH and CsI experiments were also made at low and high relative humidities, see Table 8. For both materials the ratio of low to high relative humidity aerosol mass concentration half life is about 2. This is half compared with the corresponding value obtained in the NaOH experiments. The difference is due to the density effect. The particle settling rate is proportional to the square of the aerodynamic size. The aerodynamic size on the other hand is proportional to the particle size and to the square root of its density. CsOH and CsI have higher densities than NaOH and thus during condensation their AMMD do not increase as much as the AMMD of NaOH aerosol. The value 2 for the CsOH ratio in low and high RH experiments agrees with the measured dry and wet CsOH size distributions.

CsOH absorbs water at all RHs, but CsI starts to absorb water as the RH exeeds about 91 %. Consequently, CsI mass concentration behaves similarly at RHs 27 % and 80 % (dry particles), but CsOH and NaOH behaviour is a smooth function of RH.

The half life of the silver aerosol was about 50 min, which is less than the half life for the other species. This is due to the larger aerodynamic mass median diameter (AMMD) of the high density silver aerosol. Silver behaved as a dry aerosol in all humidities and no hygroscopic effects were detected.

CsOH+Ag experiments

CsOH+Ag measurements were made at 91 % and 97 % relative humidities. Two experiments were made at 91 % RH using different material ratios between CsOH and Ag.

Silver and cesium hydroxide aerosols were generated and injected into the AHMED vessel simultaneously. The elemental particle size distribution for Ag and Cs was measured at wet conditions with four low pressure impactors installed inside the vessel.CsOH and Ag behaved as single aerosol species and no co-agglomeration was observed. This is mainly due to the relatively low concentrations used in the experiments. At higher concentrations co-agglomeration may be more important. The results show, however, that co-agglomeration is not a significant process at number concentrations below 10^5 /cm^3. Multicomponent aerosol modelling may be therefore needed for accurate containment aerosol modelling.

It was possible to determine experimental aerodynamic mass median diameters (AMMD) values from the impactor samples. The measured AMMDs can be compared to the corresponding equilibrium values calculated from the literature data. If the van't Hoff factor is known, the equilibrium AMMD can calculated from the following eq. (Jokiniemi 1990a):

$$AMMD = AMMD_0 \left\{ \left(\frac{\rho_p}{\rho_{p,0}} \right)^{1/2} \left[\frac{iM_W}{M_S (1/RH - 1)} \right] + 1 \right\}^{1/3} \tag{1}$$

where

i	is van’t Hoff factor
M_S	is molecular weight of CsOH
M_W	is molecular weight of water
$\rho_{p,0}$	is dry density
ρ_p	is particle density
ρ_w	is water density
AMMD	is aerodynamic mass median diameter
RH	is relative humidity

The van't Hoff factor for CsOH is reported only down to 95.61 % RH (Robinson 1970). Reliable values can be obtained by a semi-empirical method (Meissner 1980 and Jokiniemi 1990b) down to 90 % RH. The measured AMMD values and those calculated by equation are presented in Table 9.

Table 9. Comparison of AHMED results and literature/calculated data on particle equilibrium AMMD.

	AHMED	Calculated
RH	AMMD(eq)	AMMD(eq)
97.0 %	3.4 µm	3.55 µm
91.5 %	2.6 µm	2.77 µm
91.0 %	2.7 µm	2.74 µm

The values agree within the measurement accuracy. The comparison supports the AHMED measurements and the method to calculate equilibrium particle size from Eq. 1 using van't Hoff factors from literature data.

The AHMED experiments fill the gap that previously existed on knowledge of hygroscopic aerosols. The aerosol mass concentration could be monitored on-line and the wet aerosol size distribution was measured directly in containment conditions. As the steam condensation at supersaturated conditions is also well known (e.g. Wagner 1995), it can be concluded that the data base for steam condensation on aerosol particles is now sufficient for containment code validation.

International AHMED code comparison exercise

OECD/CSNI PWG4 accepted AHMED NaOH tests as a basis for a code comparison exercise (Mäkynen & Jokiniemi 1995). The objective was to test the containment aerosol codes for well defined boundary conditions prior to the more complex VANAM standard problem, in which aerosol behaviour was evaluated in a large scale multicompartment facility. VTT Energy acted as the host organization of the AHMED exercise.

A total 10 calculations were submitted, by 6 institutes using 6 codes. The participant list is given in Table 10. The codes represent a wide variety of containment codes used for severe accident consequence assessments.

Table 10. AHMED Code Comparison Exercise Participation.

Code	Organisation
MELCOR 1.8.3	VTT Energy, Finland
MELCOR 1.8.2	ECN; Netherlands
MELCOR 1.8.3	ENEA, Italy
IDRA 4.1 [*)]	ENEA, Italy
CONTAIN 1.12	JRC, Italy
CONTAIN 1.12	ECN, Netherlands
CONTAIN 1.12	VTT Energy, Finland
FIPLOC-MI 2.0	GRS; Germany
MACRES	NUPEC, Japan
NAUAHYGROS 1.1	VTT Energy, Finland

[*)] two sets of calculations were submitted: One with and one without hygroscopic aerosol modelling

The codes used can be divided into three groups according to aerosol modelling principles:

1) Codes including description of hygroscopic particles:
CONTAIN 1.12, FIPLOC-MI, NAUAHYGROS 1.1 and second IDRA results.

2) Codes which do not model the growth of hygroscopic particles:
MELCOR 1.8.2, MELCOR 1.8.3, and first IDRA results.

3) The MACRES code.

The codes in the first group described rather well hygroscopic NaOH aerosol behaviour in all relative humidities. The aerosol model used in CONTAIN and FIPLOC does not calculate the change of particle density as NaOH absorbs water. The codes can hence not correctly predict the effects due to changes in particle density. This effect is not very dramatic in the AHMED tests, however. CONTAIN and FIPLOC calculate the gas viscosity from dry air viscosity data. The effect of this simplification is small at high humidities and low temperatures, which was the case in AHMED experiments. The effect becomes significant at high temperatures and atmospheric pressures when the RH is high. At higher pressures the viscosity effect is not important, which is usually the case in containment. It can be concluded that the simplification made in CONTAIN and FIPLOC does not cause major errors in aerosol calculations.

The "dry" codes belonging to the second group overestimated the aerosol mass concentration in the test vessel atmosphere. Sensitivity calculations carried out with the NAUAHYGROS code proved that an equilibrium model can be applied up to a RH as high as 99 %, because particles rapidly reach their equilibrium size.

The results of the MACRES code were not realistic. The hygroscopicity model used in the MACRES code was improved based on the AHMED experiments and the renewed MACRES calculations agree rather well with the experimental results.

3.4.5 Experimental studies on LWR containment aerosol behaviour at VICTORIA facility

The general objective of aerosol experiments in the VICTORIA facility is to continue validation of the containment aerosol models used in the nuclear reactor accident codes. The final goal is to confirm that containment aerosol codes are able to correctly calculate the behaviour of radioactive aerosols in non-homogeneous multicompartment containments.

The VICTORIA test facility is a scale model of the Loviisa NPP ice condenser containment, having a linear scale 1:15 and volume scale 1:3375. The VICTORIA aerosol tests will be an extension from single compartment tests to multicompartment non-homogeneous T-H conditions. Large scale experiments have indicated that large local differences may exist in the temperatures, flow fields and relative humidities. This variation significantly affects the containment aerosol behaviour.

For the first tests the VICTORIA system geometry has been simplified from the original containment configuration. The vessel has been divided into two compartments: upper and lower compartment. Two small spherical holes, diameter 125 mm, perforate the plate dividing the compartments on opposite sides near the vessel vertical wall.

Soluble aerosol will be generated using two opposite jet atomisers. High temperature Laminar Entrainment Flow Reactor (LEFR) is used to generate inert aerosol. By choosing the optimal precursor materials and LEFR operation at high temperature this method can be applied to generate non-soluble aerosol materials.

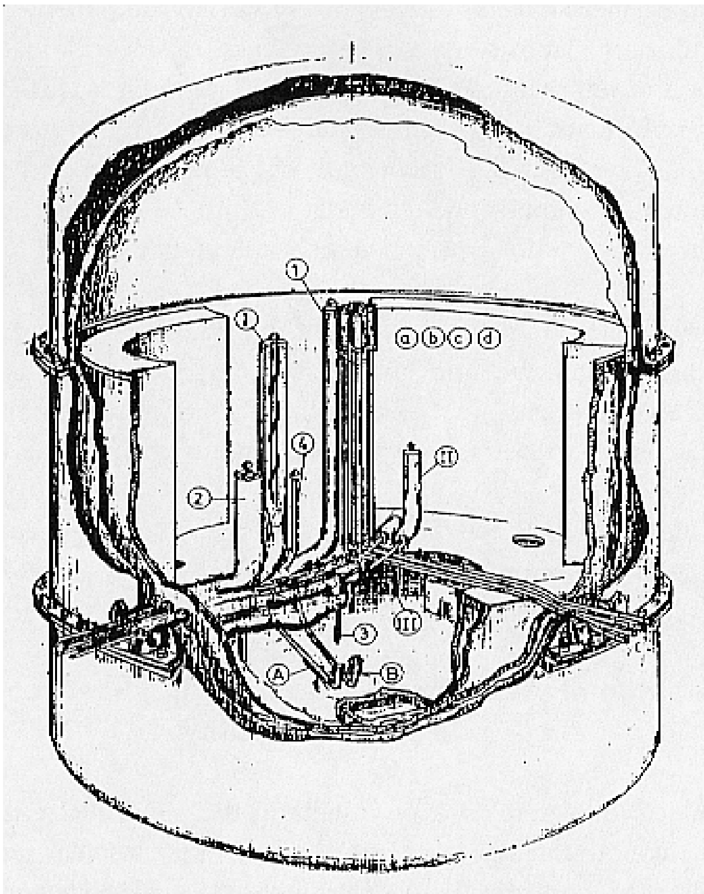


Figure 57. A schematic of the aerosol sampling lines of the VICTORIA test facility (1, 2 and 3 are TEOM and CNC lines. a, b, c and d are impactors. A and B are steam and aerosol input lines).

State-of-the-art aerosol measurement systems similar to those used in the AHMED tests will be utilized. Location of the instruments is shown in Fig. 57. The thermal hydraulic data sampling system measures and records flow rates, pressures, temperatures and RH-values in the system. Seven tests have been currently made with the modified VICTORIA facility. Four of them have been thermal hydraulic tests for system calibration, the first aerosol tests have been conducted with NaOH and CsOH as aerosol species.

3.4.6 Containment model of APROS

Plant modifications for severe accidents, and new SAM operating procedures are to be completed within a couple of years. A simulation tool for training of operators and technical support personnel for severe accidents is needed when the new systems are introduced. The training tool will be developed by implementing selected severe accident models into the general APROS - Advanced Process Simulator - environment. Development will be focused to model the severe accident management actions planned for the Loviisa plant in the first stage. Outline of the future system has been presented in (Sairanen 1993).

Model development conducted within the VAHTI-project was begun by improving the APROS containment model to describe the physical conditions and accident management hardware relevant for severe accidents. The most important phenomena modeled are:

- steam/air/non-condensable gas mixture thermodynamics
- condensation and evaporation on the heat structures and water droplets
- heat transfer to and heat conduction in the heat structures
- intercell flows

The engineered safety features include:

- the internal and external spray systems
- ice condensers in case of the Loviisa plant
- suppression pool system
- hydrogen control devices

An important feature of the simulation tool is the calculation speed. Simplifications and compromises have been therefore done in some sub-models to increase computational speed. Many sub-models e.g. the internal spray model include however sophisticated features to maintain the necessary level of accuracy needed in containment safety analysis

The renewed APROS containment model includes several modifications and improvements. All models of the older code version have been reviewed, and some of them have been completely reprogrammed. The new models developed for the current code version include non-condensable gas behaviour (others than air), its effects on mass transfer phenomena, and buoyancy effect.

Main sub-models of new APROS containment model are shown in Table 11.

The containment model is based on the common compartment formulation. The geometry is described by control volumes (nodes) connected with flow paths (branches). The containment can be arbitrarily divided into several nodes connected with both gas and water branches, enabling description of various containment geometry by a flexible way.

The model applies a lumped parameter approach: the gas region of each node is assumed to be perfectly mixed consisting of homogeneous mixture of non-condensable gases and vapour. In the current code version five non-condensable gases can be modelled: air, oxygen, nitrogen, hydrogen, and helium. Effects of non-condensable gases on heat and mass transfer parameters are taken into account.

Table 11. Status of the sub-models of APROS containment model.

Model	Yes	No	Under development
Steam/non-condensable gas mixture thermodynamics	x		
Water droplet (fog) phase	x		
Intercell gas flow	x		
Buoyancy effect	x		
Intercell water flow	x		
Intercell fog flow		x	
Heat and mass transfer			
between gas region and heat structures,	x		
between gas region and water droplets,	x		
between gas region and water pool	x		
Heat transfer between water pool and heat structures	x		
Pool boiling heat transfer		x	
Heat conduction inside heat structures	x		
Explicit sources and sinks vapour, water, non-condensable gases, dry energy	x		
Ice condenser	x		
Internal spray system	x		
External spray system	x		
Water pool	x		
Suppression pool with blowdown pipes	x		
Aerosol behaviour		x	
Fission product behaviour		x	
Cavity phenomena (MCCI)		x	
Fan coolers		x	
Gas combustion			x
Thermal radiation		x	
Hydrogen recombiners			x

Each node may also include a water droplet (fog) phase. Water droplets can be formed from unbalanced steam or as a result of direct water injection to the droplet phase. A containment node may also include a water pool (sump). The pool water is assumed to be single-phase liquid, but a separate iteration is used for the surface interface temperature, due to the strong effect on the combined heat and mass transfer calculation.

One-dimensional heat conduction is calculated in cartesian, cylindrical or spherical coordinate systems. Heat structures can be divided into several calculation layer to define the temperature distribution inside the structures. Each node can contain an arbitrary number of heat structures connected to the gas region or water pool.

Atmospheric convection heat transfer is a sum of sensible and latent heat transfer. A heat structure can be internal to a node or it can be connected to two adjacent containment nodes.

The user can choose between three basic formulations to model the structure-to-atmosphere heat transfer. Available options are:

1) the Uchida correlation
2) a condensation model for turbulent forced and natural convection
3) a mass diffusion model based on the Ackermann's approximate corrections.

The forced convection model is based on the Reynolds analogy for the heat and momentum transfer. Natural convection model is derived from the Reynolds-Colburn analogy. Different combinations of forced and natural convection Nusselt's numbers can be used depending on the input specification by the user.

Convective velocity needed in heat transfer calculation can be specified as a time-dependent input value for each structure. The velocity can also be calculated internally by the code using a simplified method based on the inlet and outlet flows of the node branches.

External sources into the containment can be specified by input tables. If a blowdown table is used, also water flashing is calculated. Steam, water, non-condensable gases, and dry energy can be injected into the containment system. The substances can also be removed from the containment system. Sources can also be obtained directly from the primary or secondary system of the APROS thermal hydraulics model.

The ice condenser model assumes that ice is contained in vertical buckets. Heat and mass transfer between the gas region and ice surface is treated analogous to heat structures. Ice heat transfer differs in some respects from solid surfaces. Therefore, the user has a possibility to modify the heat transfer coefficient to simulate turbulence inside the ice condenser, for example. Conduction inside the ice bed is not modelled. Optional heat transfer correlations are Uchida correlation or mass diffusion theory based on Ackermann's approximate corrections. A time-dependent heat transfer coefficient can also be specified by the user. Melting in the ice buckets proceeds in either vertical or horizontal direction. The cross-sectional flow area between the ice buckets is not assumed to change as the ice melts. New heat transfer area and new ice mass are updated in every time step during the melting. Energy transfered from the gas increases first the ice temperature to its melting point. After melting, the melted water warms up to a user-specified outflow temperature and drains to a sump.

The internal spray system model evaluates the heat and mass transfer between the node gas region and falling spray droplets. The model takes into account changes of droplet size due to steam condensation or evaporation, influence of non-condensable gases on the heat and mass transfer, and drag forces. Droplet temperature is calculated during the falling and the physical properties are updated due to changes in temperature. Droplets have an user-specified initial velocity and angle. Both vertical and horizontal velocities are calculated in each time step during droplet falling. Droplet interface temperature is separately iterated due to its strong effect on the combined heat and mass transfer. Heat conduction within the droplets is modelled by a simplified method. A maximum of five droplet size classes can be taken into account.

The Loviisa containment has been equipped with an external spray system for long term pressure control. The APROS external spray model includes natural convection and condensation phenomena on an inner surface of a heat structure, coupled with one dimensional heat balance calculation of the water film on the outer side of structure. As a result, temperature, thickness, velocity, and heat transfer coefficient of the external water film are calculated as a function of angular position of the containment dome. Finally, the total energy extracted from the containment is evaluated.

Each containment node can contain a water pool, and water can overflow from one pool into another. Heat and mass transfer between the pool and gas region is implicitly calculated. Pool surface temperature is separately iterated from mean temperature due to its strong effect on the combined heat and mass transfer. The pool boiling model presently available is suitable for simple simulation cases only.

Model of a suppression pool system consists of a combination of a vent pipe system and a water pool system typically used in BWR containments. Dimensions of blowdown pipes can be arbitrary specified by the user. Several suppression pools can be connected with intercell branches. Movement and acceleration of water plug inside a blowdown pipe is calculated based on the pressure difference between the upper and lower end of the pipes. The user can specify the fraction of gas flow, which will interact with the pool water. The user can also affect the outflow temperature of gases from the suppression pool.

The branch is a flow path between two adjacent containment nodes. Six different types of gas branches are modeled:

1) An open orifice
2) A flap type door. Only one direction flow is allowed when a user specified pressure difference is reached. When the pressure difference decrease below the user specified value, the door will close
3) A rupture disc. The branch opens when the pressure difference specified for the disc is reached. Once opened the branch remains open
4) Vent pipes of the suppression pool system
5) Lower inlet doors of the ice condenser

6) Intermediate and top deck doors of the ice condenser.

Two types of water branches are modeled:

1) An open orifice. Water mass flow rate is calculated according to the node pressure differences and the hydrostatic pressures in the branch inlet and outlet.
2) A simple overflow model. All water above the branch elevation will be transferred to the sump in the output node.

References for section 3.4

Anttila, M. 1991. Possibility of recriticality during a severe accident in the Olkiluoto (TVO) reactor. Report STUK-YTO-TR-26. February 1991. 28 p. (in Finnish).

Haschke, D. and Schöck, W. 1988. Results of the DEMONA aerosol removal demonstration experiments. In: Water-cooled reactor aerosol code evaluation and uncertainty assessment, Proceedings of a workshop held in Brussels (Belgium) from 9 to 11 September 1987, EUR 11351 EN. Edited by E. della Loggia and J. Royen, 1988. Pp. 88 - 99.

Højerup, F., Miettinen, J. Nilsson, L., Puska, E. K., Sjövall, H., Anttila, M. & Lindholm, I. 1997. On Recriticality During Reflooding of a Degraded BWR Core. Report NKS/RAK-2(97)TR-A3. 68 p.

Ikonen, K. 1996. Creep in Thermo-Elastic-Plastic Deformation and Stress Analysis. VTT Energy, Nuclear Energy, Technical Report VAHTI-8/96. 48 p.

Jokiniemi, J. 1990a, The effect of airborne hygroscopic matter on aerosol behavior in severe nuclear power plant accidents. Espoo: Technical Research Centre of Finland, Publications 59. 36 p. + app. 91 p. (Dissertation).

Jokiniemi, J. 1990b, The Effect of Selected Binary and Mixed Solutions on Steam Condensation and Aerosol Behavior in Containment. Aerosol Science and Tech. 13(2).

Karppinen, I. 1996a. Interfacial friction test with the APROS 4.04 6-equation model. VTT Energy, Nuclear Energy. Technical Report VAHTI-10/96.

Karppinen, I. 1996b. Simulation of the IVO full-scale loop seal experiment with APROS: VTT Energy, Nuclear Energy. Technical Report VAHTI-12/96.

Karppinen, I. 1996c. Simulation of the LOTUS air/water vertical annular flow experiment with APROS 2.11. VTT Energy, Nuclear Energy. Technical Report VAHTI-13/96.

Lindholm, I., Nilsson, L. Pekkarinen, E. & Sjövall, H. 1995a. Coolability of Degraded Core Under Reflooding Conditions in Nordic Boiling Water Reactors. Report NKS/RAK-2(95)TR-A1. September 1995. 73 p.

Lindholm, I., Nilsson, L. Pekkarinen, E. & Sjövall, H. 1995b. Analysis of Core Coolability Under Reflooding Conditions for Nordic Boiling Water Reactors. Proc. International ENS Topical Meeting on Safety of Operating Nuclear Power Plants. September 24 - 27, 1995. Budapest, Hungary.

Lindholm, I. 1996. On Core Melt Behaviour in the Reactor Pressure Vessel Lower Head in TVO Boiling Water Reactor. Technical Report VAHTI-9/96. 30 p.

Mattila, L. & Vanttola, T. 1995. YKÄ Research Programme on Nuclear Power Plant Systems Behaviour and Operational Aspects of Safety 1990-1994. Final Report. Ministry of Trade and Industry, Helsinki. 261 p. (Ministry of Trade and Industry Studies and Reports 98/1995). ISBN 951-739-023-8.

Meissner, H. 1980. Prediction of Activity Coefficients of Strong Electrolytes in Aqueous Systems. ACS Series 133:495-511.

Mosteller R. D. & Rahn F. J. 1995, Monte Carlo Calculations for Recriticality During the Reflood Phase of a Severe Accident in a Boiling Water Reactor. Nuclear Technology, Vol. 110 (May 1995), pp. 168 - 180.

Mäkynen, J., Jokiniemi, J. & Kauppinen, E. 1994. Experimental Studies on Hygroscopic Aerosol Behaviour in LWR Containment Conditions. In: J. Aerosol Sci 25, S1. The 1994 European Aerosol Conference Proceedings, Blois, France. May 30 - June 2, 1994. pp. 247 - 248.

Mäkynen, J. & Jokiniemi, J. (Ed.) 1995. AHMED code comparison exercise. CSNI/PWG4/FPC Comparison Report, Le Seine Saint Germain, France, October 1995. 80 p.

Okkonen T., Hyvärinen J. & Haule K. 1993. Safety Issues Related to Fuel-Coolant Interactions in BWRs. Proc. Of CSNI Specialist Meeting on Fuel-Coolant Interactions. January 5 - 8, 1993, Santa Barbara, USA. NUREG/CP-0127, NEA/CSNI/R(93)8, pp. 296 - 307.

Owen, D. G., Hewitt, G. F. & Bott, T. R. 1985. Equilibrium annular flows at high mass fluxes; Data and interpretation. PHC Physico Chemical Hydrodynamics. Vol. 6, No ½, pp. 115 - 131.

RELAP Development Team, 1995. RELAP5/MOD3 Code Manual, Idaho National Engineering Laboratory, Lockheed-Martin Idaho Technologies Company, NUREG/CR-5535, INEL-95/0174, Vol. 1.

Robinson, R. A. & Stokes, R. H. 1970, Electrolyte solutions, 2nd Rev. Ed. London, Butterworhs.

Sabathier, F., Zoulalian, A., Albiol, T. & Layly, V. 1994. The growth of hygroscopic aerosol in humid atmosphere. In: J. Aerosol Sci 25, S1. The 1994 European Aerosol Conference Proceedings, Blois, France May 30 - June 2 1994. Pp. 273 - 274.

Sairanen, R. 1993. A training simulator to support the Loviisa VVER-440 severe accident management programme. Proc. OECD/NEA/SESAM Specialist Meeting on Operator Aids for Severe Accidents Management and Training. Halden, Norway 8-19 June 1993. Halden: Institutt for Energiteknikk. p. 219-234. (NEA/CSNI/R(93)9).

Scott, W. B. et al. 1990. Recriticality in a BWR Following a Core Damage Event. NUREG/CR-5653, PNL-7476.

Shamoun B. I. & Witt R. J. 1994. Parametric Study of Recriticality in a Boiling Water Reactor Severe Accident. Nuclear Technology, Vol. 107 (Aug. 1994), pp. 227 - 235.

Tuomisto, H & Kajanto, P. 1988. Two-Phase Flow in a Full-Scale Loop Seal Facility. Nuclear Engineering and Design 107, pp. 295 - 305.

Tuunanen, J. & Kouhia, J. 1995. Modifications of the PACTEL Test Facility for the Gravity Driven Emergency Cooling Tests (GDE-11, -12, -13 and -14). VTT Energy, TEKOJA11/95.

Wagner, P. 1995. Condensation processes in aerosols, J. Aerosol Sci. Vol. 26 S 1, pp. 203 - 204.

Wilson, J. H., Wright, A. L. & Arwood, P. C. 1988. LACE aerosol code-comparison activities at Oak Ridge Laboratory. In: Water-cooled reactor aerosol code evaluation and uncertainty assessment, Proceedings of a workshop held in Brussels (Belgium), 9 - 11 September 1987, EUR 11351 EN. Edited by E. della Loggia and J. Royen, 1988. Pp. 9 - 20.

3.5. Risk and reliability analyses

Risk and reliability analyses have become increasingly important tools in the control of nuclear safety. Probabilistic safety analysis (PSA) is currently used as a regulatory licensing tool and as background for utilities' major safety improvements. This is due to the fact that it gives means to model the uncertainty involved in nuclear power plant safety. This uncertainty is often expressed as the probability of hazardous consequences - the worst consequence being an extensive core damage studied in PSA.

PSA attempts to comprehensively identify all important risk contributors, compare them with each other, assess the safety level and suggest improvements based on findings. The strength of PSA is that it is capable of providing the decision makers with numerical estimates of risks. This makes the decision making far more easier than the comparison of purely qualitative results. PSA is the only comprehensive tool that compactly attempts to include all the important risk contributors in its scope.

Despite the demonstrated strengths of PSA, there are some features that have reduced its uses in the past. For example, the PSA scope has been rather limited to the power operation and process internal events (transients and LOCAs). Only lately, areas such as shutdown and severe accidents have been included in PSA models in many countries.

Problems related to modelling are, e.g., that static fault and event tree models are commonly used in PSA to model dynamic event sequences. Furthermore, there is a great variety of different techniques for human reliability assessment (HRA) giving varying results. Finally, PSA gives only results with regard to core damage risk, and in decision making, also other criteria such as power production availability and maintainability may become important.

In the project Reliability and Risk Analyses (LURI), these limitations and shortcomings are under investigation. Decision making connected to nuclear risks and its support methods are the main themes of the project. The project also connects the use of probabilistic methods to themes like uncertainties involved in man-machine systems, and safety critical organisations. Joint projects between different disciplines give a wider viewpoint to the whole safety issue.

The project continues the work of ‘Living probabilistic safety assessment and risk decision-making’, belonging to the Programme YKÄ under 1991 - 1994. The themes of that project were NPP performance indicators, living PSA methods and risk decision making test cases.

3.5.1 Treatment of uncertainties in risk analysis by using expert judgement

Some issues of PSA are subject to various uncertainties. In level 1 PSA, the uncertainties are often related to the lack of sufficient statistical data concerning the failures of components, frequencies of initiating events and human errors. In level 2 PSA, the uncertainties are connected with complex phenomena, which cannot be analysed by using largely accepted or established methods and which cannot be studied empirically. An example of such phenomena is steam explosion. In PSA, the uncertainties which cannot be analysed with validated models and empirically, are often treated by applying expert judgements (see Cooke, 1991, and NUREG-1150, 1989). Another way to take this issue into account is to treat the model uncertainty explicitly. At the outset of the LURI-project the issue of uncertainties was considered from both the model uncertainty and expert judgement point of views.

The <u>model uncertainty</u> was dealt within a project in cooperation with Finnish Center of Radiation Protection and Nuclear safety (STUK). Usually, the uncertainties of risk model parameter values are described by probability distributions and the uncertainty is propagated through the whole risk model. In addition to the parameter uncertainties, the assumptions behind the risk models may be based on insufficient experimental observations and the models themselves may not be exact descriptions of the phenomena under analysis.

In the project, some approaches to model and quantify the model uncertainties were discussed. The emphasis is on so called mixture models, which have been applied in PSAs. Some of the possible disadvantages of the mixture model were addressed. These models apply rather well in most cases in PSA, where the modelling uncertainties are rather well defined: usually the alternative models are based on a small number of clearly defined different assumptions, one of which is the true one. For this type of phenomena the mixture approach is fully applicable. One advantage of the mixture models is that they may be easily incorporated the conventional uncertainty analysis of level 1 PSA. In addition to quantitative analyses, also qualitative analysis is discussed shortly. To illustrate the models, two case studies were carried out.

In the case of complex physical phenomena, a model consisting of set of exclusive and exhaustive assumptions may not give the right picture. This may be due to the fact that the behaviour of the system must be studied for a long period of time and there is a considerable uncertainty. In those cases, more general approaches should be used; one possibility is to interpret the models as a kind of expert judgements and, then, apply Bayesian inference. In practice, one of the basic features of this general model, the likelihood function, can hardly be determined. Despite of this practical difficulty, it is important to notice the insufficiencies of the mixture model approaches, and to develop more general models.

The study revisited the fact that it is not always possible to make a distinction between model and parameter uncertainty. The analysis on how the uncertainty is reduced, when additional evidence is obtained, revealed the fact that due to the larger dimensionality of the "models of model uncertainty", the uncertainty is decreased more slowly than in the case of parameter uncertainty.

To illustrate the treatment of model uncertainties we considered two simple case studies. In both examples, the analysis is based on simple mixture models, which are observed to apply in PSA analyses.

In the first study, the failure intensity of a component was modelled when the functional form of the intensity is unknown. The approach based on various mixture models. The main issue of the modelling uncertainty in this example was the increase of the model dimension, which is in the most general situation infinite. By adopting a Bayesian approach, the statistical inference is possible even for nonparametric (infinite dimensional) models.

The second case study dealt with the modelling of human error initiators in a low power and shutdown PSA. One of the basic uncertainties in that case study was described by an error category model, which could be easily quantified. Another model uncertainty, which we identified, but not quantified, was the uncertain decomposition of the human errors involved. The Boolean decompositions consisted mainly of Boolean products, which causes the fact that the probability of the whole event is the product of the probabilities of the sub-events. The decomposition itself may be wrong, and furthermore, the evidence (in form of expert judgements) on the sub-events may be missing.

One of the basic principles of probability modelling is that the model elements must, in some way, correspond to the evidence available, which may not be the case in our case study. If this kind of contradiction between model and the evidence is possible, one should not decompose the events. If decomposition is however made, one way to interpret the probabilities of the sub-events is to think them as some kind of performance shaping factors (PSFs).

PSAs are probabilistic analyses dealing with uncertainties. Thus, it is natural to include a probabilistic analysis of modelling uncertainties to the framework of PSA. When a quantitative perspective is adopted, the modelers are required to express their uncertainty in an exact manner following the rules of probability. Further, to model uncertain phenomena, the modelers must identify the assumptions of the models and evaluate their significance. This is basically the qualitative step in the analysis process, and it should be made and documented systematically. Complex mathematical formulations of the model uncertainty may hide the importance of qualitative analysis, which is the basis of any risk analysis.

The methodology for model uncertainty analysis is still under development, and more experience on various approaches is needed. The issue is connected with other problematic areas of risk assessment, such as the use of expert judgement. In this situation, it may be too early to require quantitative model uncertainty assessments to be included in plant specific risk analyses. To obtain further experience, various benchmark studies on clearly defined modelling problems may be organised.

On of the most extensive and systematic application of expert judgements is described in the NUREG-1150-document. The methodology applied in NUREG-1150-project consists of several steps including of 1) selection of issues which require expert judgements, 2) selection of experts, 3) training of experts, 4) eliciting expert judgements and 5) combining the judgements. The application of the original NUREG-1150 methodology requires rather much resources, and it is not feasible to apply it in most extensive form for all issues requiring expert judgements.

In the LURI-project, also a new methodology for using and eliciting expert judgements was developed. The methodology has its background in the NUREG-1150: the selection of issues for expert judgement, selection and training of experts and the elicitation process are simplified versions of the NUREG-1150-method. However, the experts' judgements are combined by using a Bayesian framework (see e.g. Pulkkinen 1994, Pulkkinen & Pyy 1996). The application of the Bayesian framework makes it possible to elicit the judgements also in more general format. For example, it is possible to apply ordinal comparisons, direct estimates or probability distributions parametrized by selected fractiles. The expert methodology is outlined in the following.

After selection of the issues and experts, the experts are trained for their task. In order to make sure that the experts interpret their task in assessing unknown parameters and to help the expert to encode their assessments in probabilistic terms a short introduction to the probabilistic expert judgement methodology is needed. Furthermore, one objective of the introduction is to ensure that the experts accept the methodology and understand its principles. The introduction is given during a training session.

The training includes discussions on the concepts of probability theory and statistics. If necessary, the concepts are illustrated by simple examples. The expert judgement methodology is presented and it is applied to simple examples. In addition to the methodology, the significance of the experts assessments with respect to the issue under analysis is discussed. The use of the results is described and the necessary information about the issue is given. The models describing the phenomena are presented and interpretations of their parameters are given.

As the next step, the judgements are elicited from the experts. The elicitation of experts estimates may be a difficult task. In some cases, it is enough to ask the experts to give estimates for the unknown variables, sometimes they are also asked to assess their uncertainty about the estimates - or to give a whole probability distribution for the variables. The elicitation process depends both on the phenomena under analysis and on the form of the discussions between the experts. In the methodology developed in the LURI project, the experts are allowed to discuss about the issue, but numerical estimates are not deeply discussed during the session. During the discussion, the assessment task is clarified and the models and the decomposition of the phenomena are revisited. Further, the nature of probability distributions applicable into the case under consideration is discussed. The normative experts leads the discussion.

The experts give their assessments individually after the discussions. The assessments are written on a form, which is developed specifically for the issue, and they are discussed with the normative expert. Since it is possible that the unknown random variables are correlated, the issue of dependence is considered and taken into account. If possible, computer tools are used in the individual elicitation sessions. Depending on the case, the experts assessments may be direct estimates, estimates with uncertainty bounds (e.g. .05-, .50-, .95-fractiles), or continuous or discrete probability distributions. In addition to the above assessment, more incomplete information, such as pairwise comparisons may be asked.

The objective of the expert judgements is to collect data on uncertain phenomena and to take into account and measure the uncertainty. This is realised by combining the judgements, which is the next step of the method by a Bayesian approach. The features of our model are the prior distribution for the variables and the model for experts assessments. If there is not much prior information on the variables, then the prior distribution is chosen so that it doesn't have much effect on the posterior distribution. It must be noticed that the form of the prior must be specified separately for each case.

Fig. 58 illustrates the expert judgement model developed in the LURI project. In the model, X is the unknown variable of interest. The variables $\boldsymbol{Y_i} = (Y_{j0.05}, Y_{j0.50}, Y_{j0.95})$ are the percentiles given by the experts, and Θ_{jl}, μ and σ are the parameters of the model. Also the error term ù in the individual assessment was taken into account. In our case, the model is based on transformations of Gaussian distributions, and the computation is based on Markov Chain Monte Carlo methods (e.g. the Gibbs sampler).

The methodology developed in LURI was applied in the Benchmark exercise organised by the European Commission as a concerted action within the context of the IV EC Framework programme (see Cojazzi et al. 1996). In this Benchmark exercise, different methodologies were applied into the analysis of fuel-coolant interaction experiment. The expert team consisted of a normative expert and four substance experts. As a part of the Benchmark exercise, reports describing the method and the applications of expert judgements in Finland were written (see Pulkkinen & Virolainen 1996, Pulkkinen & Holmberg 1996) The project is still running, and further results will be published in the near future.

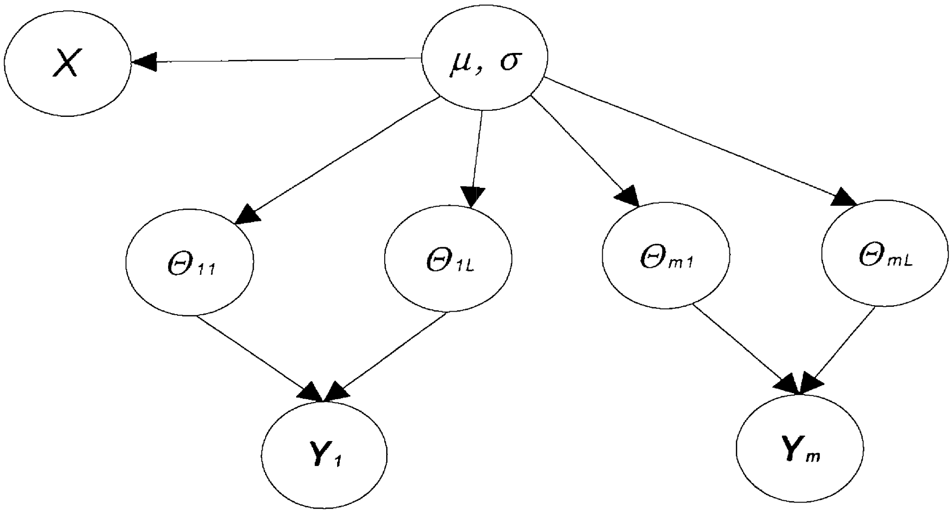


Figure 58. Bayes network for the expert judgement model.

3.5.2 Methods for decision support

Decision analysis applies methods of systems analysis and operations research in supporting important decisions. The two essential modelling problems in risk management context is the treatment of uncertainties and treatment of multiple objectives. In the LURI-project, both aspects are studied in order to develop the decision analysis methodology for operational safety management of an NPP.

As a part of the project, international applications of decision analysis in nuclear safety were surveyed. The survey was based on selected published reports and a questionnaire sent to the members of the principal working group on risk analysis (PWG5) of OECD/NEA/CSNI (Holmberg & Pulkkinen 1995). Application areas can be divided into: use of probabilistic safety criteria, operational safety management, nuclear waste management and emergency management.

The experience from the decision analysis methodology has been rather positive. Analytical thinking helps structuring the problem and explicating statements on uncertainties, values and preferences. The decision analysis methodology is rather mature to be applied in solution of nuclear safety issues. Although the reported international applications have mainly been research oriented, it can be expected that practical uses of decision analysis will become more common. In Table 12 the obtained results are shown.

Table 12 . Comparison of the decision analysis studies in nuclear safety field.

Application	Problem area	Purpose	Participants	Model
EPRI	Design modification			value-impact model, MAVF
Nelson and Kastenberg	Design modification	Demonstration	Researchers	value-impact model, AHP
Nordic benchmark study	Temporary plant shut-down	Benchmark	Researchers	MAVF, AHP, utility function
STUK exercise	Temporary plant shut-down, design modification	Exercise	Safety authorities, researches	MAVF
Dutch hydrogen workshop	Design modification	Decision support	Experts in nuclear safety	MAVF
U.S. national waste management	Waste management strategy		Researchers	MAVF, decision tree
Analysis of protective actions after an accident	Accident management strategy	Demonstration	Researchers	MAUF
Decision conference on countermeasures after a accident	Accident management strategy	Simulation, exercise	Officials, experts	MAVF
NRC study on Application of formal decision making methods to safety issue decisions	Design modifications, retrofitting	Evaluation of methodology	Researchers, NRC decision makers	MAUF

AHP = Analytic hierarchy process

MAVF = Multi-attribute value function

MAUF = Multi-attribute utility function

As a practical application of the methodology, a decision analysis procedure was developed for Loviisa power plant. The procedure starts with a qualitative decision analysis during which options and criteria are identified as well as options are evaluated qualitatively. In the next step, a quantification of the options is made based on a multi-attribute value and utility function model. In order to ease the analysis, the procedure includes predefined criteria sets for selected problem categories as well as trade-offs between commonly used attributes. Fig. 59 presents the objectives hierarchy for the comparison of decision alternatives.

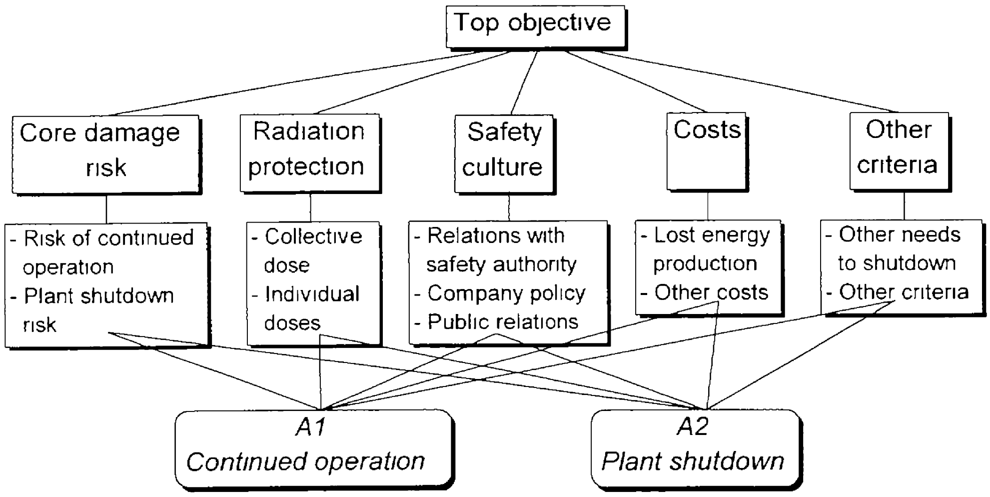


Figure 59. Decision making criteria for the comparison of NPP continued operation and shutdown options.

Theoretical studies on decision analysis, performed as a part of LURI, are related to the problem of integrating living PSA to risk management. The approach is based on the application of marked point process framework for living PSA (Arjas & Holmberg 1995, Holmberg 1996a). In this framework, risk management turns out to be a problem of optimal control of a stochastic process.

The decisions can be divided into three categories according to control methods of a point process: (1) optimal stopping, (2) intensity control, and (3) impulsive control. In terms of risk management, the optimization of the stopping time can be associated with the licensing of an NPP. By design modifications and selections between alternative components as well as operating procedures, the management can influence the average accident hazard rate of the NPP process. This is a form of intensity control. Operational decisions, like temporary shutdowns and testing, can be regarded as impulsive controls (Holmberg 1995).

The objective function of the decision model is the expected lifetime utility of the plant. The small probability of an accident, and the large negative value of its consequences, cause conceptual difficulties in an attempt to apply the expected utility theory. Holmberg (1996c) relates the concept of probabilistic safety criteria, set by safety authorities, with the definition of an utility function. A probabilistic safety criterion specifies the maximum acceptable hazard rates of various accidental consequences. Assuming that the criterion depends also on the benefit of the process to society and on the licensing time applied, these statements can be regarded as preference relations. A probabilistic safety criterion is interpreted to mean that if the accident hazard rate is higher than the maximum acceptable hazard rate, then the optimal decommissioning time of an NPP is shorter than the licensing time. This interpretation yields a limiting condition for a feasible utility function.

The optimization problem induced by the decision model is complicated, since the objective function may be discontinuous with respect to the argument and generally the expected utility cannot be evaluated analytically. Particularly, the computational difficulties are due to the self-exciting character of the stochastic process and due to the non-linear utility function. Holmberg (1995, 1996b) applies the stochastic quasi-gradient algorithm to the search for solutions.

Holmberg (1996b) applies the approach in the optimization of plant shutdown criteria and of test intervals, with increasing operating experience. Several long term strategies are analyzed and compared. In both cases, some intensity parameters of the model are assumed to be unknown, and their probability distributions are updated by the observed operating experience in a Bayesian manner. The examples demonstrate how, in principle, risk-based optimization of the operation of a hazardous process can be performed by using a living PSA model. By this way, control strategies adapted to operating experience can be studied as is needed, for instance, in risk monitoring applications. The work will continue in the form of a doctoral thesis and as a part of the VTT Automation research theme “risk based regulation”.

3.5.3 Human reliability assessment (HRA)

Human reliability is one of the problem areas of PSA. Due to this fact, the LURI-project has tried to establish new viewpoints to the topic. This has taken place both in the integrated safety analysis study and in the study of maintenance originated NPP component faults based on the maintenance data.

The integrated safety analysis study is the Finnish contribution to NKS/RAK-1.3 project “Integrated sequence analysis”. This project is a part of the Nordic four year RAK-programme (NKS 1996). The main aim of the project is to form interdisciplinary views on safety analysis, especially in cases where accident sequences involve a considerable amount of human actions.

The aim of the Finnish contribution is to integrate two complementary approaches for analysing operators’ decision making and human reliability in accident sequences. These approaches are (1) probabilistic PSA modelling and decision analytical multi-attribute utility theory (MAUT) and (2) contextual psychology (also as a part of ORINT-project).

At the project outset, as specific goals, specification of a dynamic PSA model suitable for HRA purposes and a quantitative analysis of a selected sequence by using the model were set. Also decision analytic dimensions of sequences addressed in an HRA were under a closer study. The interface between the two disciplines - reliability engineering and psychology - takes place through several reference models. As reference models, decision tables, critical information tables, different flow charts and influence diagrams are typically used (Holmberg et al. 1996).

The first application of the VTT joint approach has taken place in the analysis of the cold overpressure risk of a BWR plant. The probabilistic analysis consisted of the construction of the main probability model and its submodels. The critical point of the probabilistic modelling is to include the important human reliability contributors and various pieces of knowledge into one probability model. This means first decomposition of the sequence into smaller parts and models, then the composition of the submodels and, finally, integrating the submodels into the main calculus (main model).

The main probability model used in our first application was dynamic and expressed by a marked point process model (Holmberg et al. 1996). The submodels can be divided into physical models (e.g. pumping with a certain mass flow leads to linear level increase), decision models (e.g. preference to use certain pumps may be dependent on safety, timetable, pump availability, written procedures and common practices) and probability models for certain nodes (points including uncertainty).

The influence diagram model shows how the different above mentioned nodes (or submodels) are present in the main probability model (Fig. 60). The unwanted event is reactor overfilling leading to overpressure.

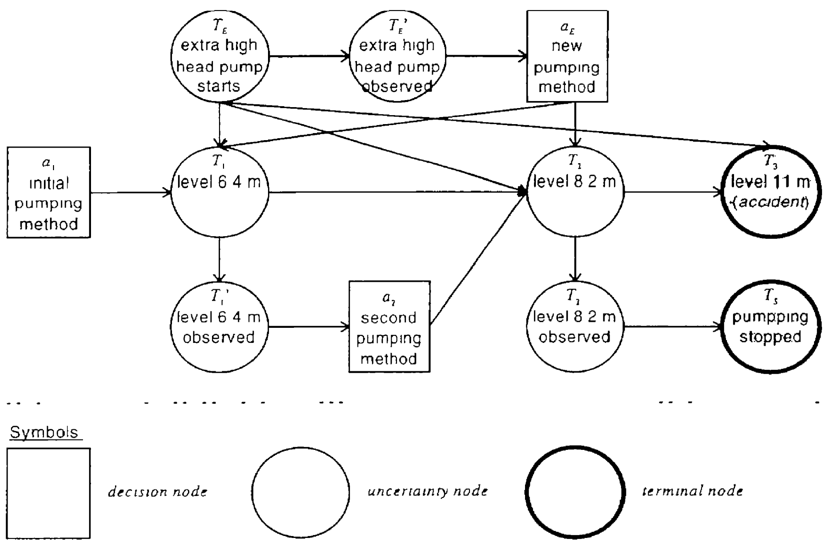


Figure. 60. Influence diagram model for overfilling and cold overpressure case.

The influence diagram in Fig. 60 was quantified by using Monte Carlo simulation. Apart from the marked point process modelling, also expert judgement was utilised in the assessment of the probability to discover the spurious start of one high head pump (Pulkkinen & Pyy 1996).

The results showed the overfilling risk to be extremely small, which supported earlier, less detailed analyses of the case (Pyy & Himanen 1993). The methodology showed its strength as a communication channel and, on probabilistic side, as a functioning tool for a detailed reliability modelling of time dependent phenomena. The next step taken is to apply the approach to an operational BWR transient and develop the methodology further.

<u>The plant history based study of test & maintenance caused component faults</u> is a joint project of the Finnish Centre of Radiation Protection and Nuclear safety (STUK) and VTT Automation. The study is based on real plant fault record data completed by utility licensee event reports and root cause analyses of detected faults and human errors.

The first aim of the study was to identify and give examples of the origin and appearance of such human originated failure mechanisms, which can penetrate the barriers of the different test and inspection processes of the plants. Another objective was to generate numerical safety indicators describing and forecasting the effectiveness of maintenance performance. The third objective was to identify decision options for further development of the quality of the maintenance and its planning. In those purposes, the distributions of following error categories were studied for both single and dependent maintenance errors: operating modes when committed, operating modes when detected, situation where observed, equipment, types of errors etc.. Similarly, observations with regard to penetration of administrative barriers and tests were made.

The study is rather unique because large quantities of plant specific maintenance records were used as the raw data, and the findings were discussed thoroughly with the plant personnel to verify them. As the first phase, altogether 4407 maintenance history fault records and 16 licensee event reports through 1992 - 94 from a NPP were analyzed. From this data, about 206 single human errors and 14 dependent ones, recorded in 250 fault reports together, were identified for the follow-up analysis. This amount corresponds about 6 % of the total number of fault records. Apart from the dependent human errors, the analysis also revealed a couple of mechanisms, where single human errors had led to multiple consequences.

With regard to single human errors, instrumentation (appt. 41 %) and electrical component (17 %) related human errors are rather frequent when compared to other component classes. This dominance seems to be even more prevailing with regard to the dependent human errors, according to the preliminary results. Similarly, qualitative (too little or too much force) and carelessness type errors seem to be frequent when compared to other error classes, like omissions normally included in the scope of PSAs.

Another interesting observation can be made with regard to the error origin and detection periods. Most of the errors - single and dependent ones - are born in outages, which was somewhat expected taking into account the amount of maintenance and modification activities at that period. In Fig. 61, the operating mode at the time of the single human error detection is shown, given that a maintenance related error was induced during an outage. Approximately 50 % of these faults are also discovered during the outage itself - the rest 50 % remain latent until the start-up or even until the power operation. The explanation, that single human errors would be negligible from the safety point of view, is not sufficient, since similar results seem to apply to dependent human errors, too. From the single errors taking place during the power operation, more than 90 % are also detected during the power operation mode.

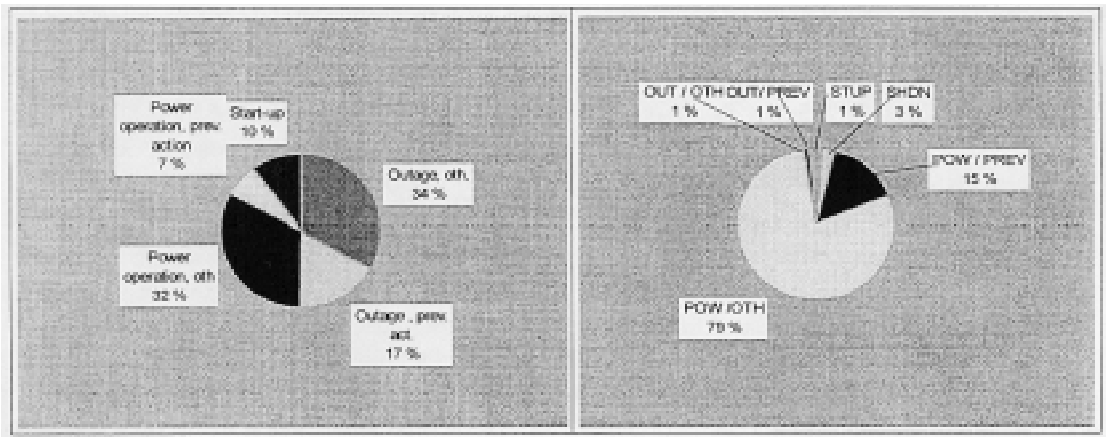


Figure 61. Plant operating mode at the time of detection of 127 single human errors stemming from outages (left) and 78 stemming from the operating period (right). OUT=outage, POW=power operation.

As common cause failures (CCFs) are, generally, the most risk inducing ones, a special effort was put to analyse them. Here, the distinction between the critical and non-critical aults, as consequences of the errors, was made. Besides, 12 cases where single human error had had consequences in multiple equipment, we identified.

These dependence mechanisms were verified by interviews with the maintenance foremen and analysed thoroughly. The analysis was rather effort consuming, since the dependences seem to be extremely difficult to classify, and it is still partly underway.

Dependent errors were found both in safety related and other systems. In critical and non-critical CCFs, plant modifications (app. 50 %) in I&C systems (app. 70 %) seem to be an important source class. Periodic testing and alarms seem to be an efficient tool to reveal dependences, although some errors were induced at the occasion of a test. Similarly, increased supervision of suppliers and improved post-installation checking might decrease the amount of dependent maintenance related errors. A general recommendation, supported by the preliminary results, is that more emphasis should be put to I&C system safety studies and testing.

The study forms a part of the Finnish contribution to OECD/NEA/CSNI/PWG1 'Human originated common cause failures', IAEA 'Human reliability data collection' and NKS/RAK 1-4 'maintenance strategies and ageing' programmes. The study will go on with further analysis of the data and, in future, data from another NPP will be analysed in order to compare the results.

Apart from the studies of integrated analysis approaches and maintenance errors, also the effect of organisation on human reliability have been addressed in a literature study illustrating various techniques (Lehtinen 1995). The studies with regard to safety critical organisations will be continued under a specific VTT Automation research theme.

As a new theme, development of HRA for extraordinary circumstances such as outage and severe accident management has been initiated. Also the effect of safety related modifications is an interesting topic from human reliability point of view. It will discussed more in section 3.5.4. The future topics of the project will include, e.g., development of human reliability analysis methodology for outage conditions.

3.5.4 Assessment of safety work and plant modification impacts on safety

Operational experience has shown that organisation, management and plant modifications are important for safety and operational performance. The research aims at identifying organisational factors by which safety and operational performance can be improved. The study has been carried out as a part of the Nordic research cooperation NKS in nuclear safety in two projects, RAK-1.1 and RAK-1.5.

<u>The RAK 1.1 project "A survey and an evaluation of safety practices"</u> aims at an identification of possible deficiencies and at an evaluation of the effectiveness of the safety. A <u>comparative study of Finnish and Swedish practices</u> in nuclear safety forms one part of the project. The comparative study was intended to provide information beyond available anecdotal evidence of differences in safety practices. Data has been collected in interviews at the plants and at the authorities in Finland and Sweden. Another part of the project is concerned with the creation of <u>a conceptual model of</u>

nuclear safety as a *map*. The map should be able to illustrate the *path* between requirements on and solutions of various safety activities.

Nuclear power in Finland and Sweden is regulated through national laws and regulations. The first atomic energy act in Sweden was adopted in 1956 and in Finland 1958. The nuclear energy acts have both in Finland and Sweden been revised several time. Laws and regulation in both Finland and Sweden empowers a national authority to act as a regulator. In Sweden this authority is split between two bodies the Swedish Nuclear Power Inspectorate (SKI) and the Swedish Radiation Protection Institute (SSI). In Finland the Finnish Centre for Radiation and Nuclear Safety (STUK) covers the mandate of both SKI and SSI.

In Finland STUK has written a comprehensive set of safety guides, YVL-guides, which take stand on various safety issues. The YVL-guides are updated regularly and go through a detailed review process before they are adopted. In Sweden, instead of using guidebooks, the American safety case based regulation model is used and reference is made to USNRC requirements. The operating license in Sweden is awarded on the basis of an application to which a FSAR is attached. When the application is accepted, the FSAR is seen as an integral part of the license and in a way as an agreement between the authority and the utility on the construction and operation of the plant (Fig. 62).

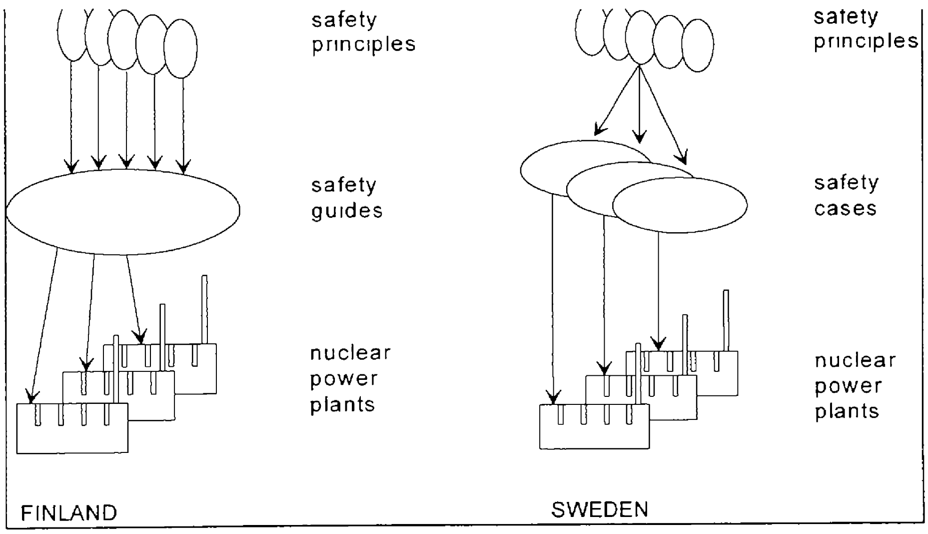


Figure 62. There is a difference in principle between Finland and Sweden in how requirements are handled in the regulatory system.

Comparing the resources of SKI with corresponding parts of STUK indicates that the STUK has absolutely more resources in spite of the fact that the nuclear power programme in Sweden is about three times as large as in Finland. This difference in the resources is also reflected in the working practices is such a way that SKI directs their inspection efforts more to the safety processes of the utility companies than to technical details. SKI is further advancing plans for a risk based regulation to help in concentrating their efforts on the most important activities.

Differences between safety practices in Finland and Sweden are traceable to historical differences in how nuclear power was introduced. For example, Sweden has its own national reactor vendor. In Finland the option to build more nuclear power plants is still open, but in Sweden a political decision puts a definite date of 2010 for the phase out of nuclear power.

A major similarity between plants in Finland and Sweden is their commercial success. All plants have been able to reach very high availability figures for many years and they can be judged to be very efficient with respect to all performance indicators. There are several reasons for this success., e.g. both Finland and Sweden have an industrial infrastructure which was able to support large high tech projects. The nuclear industry in its beginning attracted the cream of young engineers and the success of the technical constructions rely on their innovativeness. During the last years the number of university graduates with a specialisation in nuclear engineering has been decreasing both in Finland and Sweden. ABB Atom and the power companies in Sweden have initiated programs for young university graduates to be trained in various skills considered important by the industry. In Finland no such programmes are under way at the time being, but on the other hand a pool of skills is maintained by bodies such as the Technical Research Centre of Finland (VTT).

The collection of information for the comparison between Finnish and Swedish activities in nuclear safety was done on the basis of an idealised view of which activities are important. Originally the idea was expressed as a *map* of safety practices and a *path* between requirements and solutions. In the course of the project it has however been somewhat difficult to concretise this metaphor. A comprehensive picture of the safety activities can only be achieved through a combination of many different views.

The most important features of all activities are embedded in the concepts *goals*, *planning* and *feedback* (Fig. 63*)*. Safety activities should be *described* and *operational*. The description of safety activities should be accurate enough to serve as a kind of procedure for how to do the work and it should contain information on their goals. Described safety activities are operational if actual practices are in accordance with the descriptions. Inspection can then be seen as a twofold process, firstly described arrangements of the safety activities are compared with an ideal model and secondly it is verified that activities actually are performed in accordance with the descriptions.

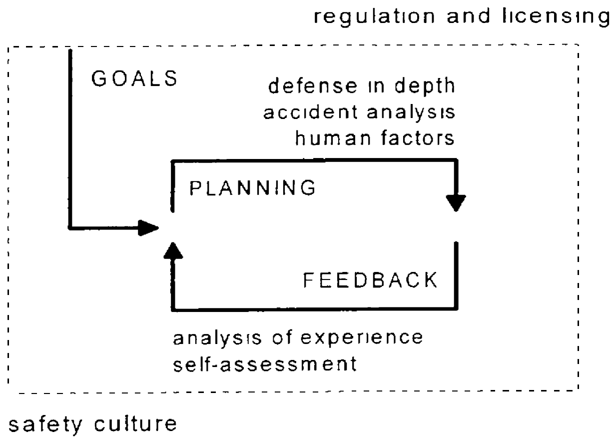


Figure 63. The general structure of the map of the safety activities in nuclear power.

Early considerations of important parts of the safety activities were used to build a structured interview form for discussions with various experts within the industry and by the safety authorities. The present more refined model was built based on these discussions. The next step in the development of the ideal model of the safety activities is to formalise the concepts to make them more consistent and accurately described. The model will also be used for an assessment of the modification activity at the nuclear power plants in Finland and Sweden.

Interviews at the nuclear power plants in Finland and Sweden will be completed to cover all sites. The map of the safety activities will be completed and described in a report. Similarities and differences between Finland and Sweden will be covered in a separate report.

In addition, several more in-depth studies have been initiated and will be reported during 1997. These studies are: a comparison of modernisations, renewal and changes at the nuclear power plants, a study of the Oskarshamn 1 restart project, a study of the handling of the clogged strainers in a larger perspective.

The Nordic project RAK 1.5 "Changes, renovation and modernisations at the nuclear power plant in Finland and Sweden" has been started during 1996. The goal of the project is to assess and compare practices by which changes, renovations and modernisations are carried out both in practice and as described in applicable instructions. Information on how the authorities and the nuclear power plant are carrying out their processes of bringing in changes to the original construction of the plants will be collected in interviews and questionnaires. This project has been triggered by the modernisation projects at the nuclear power plants in Finland and Sweden. These have been initiated to ensure undisturbed operation for additional decades and by a need to replace solutions which for various reasons are difficult to maintain. In Finland an additional motive for the modernisation projects is to increase the electric power output for the plants.

Two separate processes are crucial for an successful implementation of changes at a nuclear power plant. In the first process the power plants are according to their own routines bringing a suggestion for a modification through several phases to a final implementation. In the other process the authority is taking stand on the acceptability of the modification and the used construction processes.

The results of the study will be described in a final report which will put an effort in especially analysing differences in practices in Finland and Sweden and between the power plants.

References for section 3.5

Arjas, E. & Holmberg, J. 1995. Marked point process framework for living probabilistic safety assessment and risk follow-up. Reliability Engineering and System Safety 49, pp. 59 – 73.

Cojazzi, G., Pinola, L. & Sardella, R. 1996. A Benchmark Exercise on Expert Judgement Techniques in PSA Level 2: Design Criteria and General Framework. European Commission, Joint reserach Centre, Institute for Systems Informatica and Safety. PSA'96 International Topical Meeting on Probabilistic Safety Assessment, Park City, Utah, September 29 - October 3, 1996. 8 p.

Cooke, R.M. 1991. Experts in Uncertainty. Opinion and Subjective Probability in Science. Oxford university Press, New York. 321 p.

Holmberg, J. 1995. A decision model for the risk management of hazardous processes. Laxenburg, International Institute for Applied Systems Analysis, Report WP-95-95. 24 p.

Holmberg, J. 1996a. Risk follow-up by probabilistic safety assessment — experience from the Finnish pilot study. Reliability Engineering and System Safety 53, pp. 3-15.

Holmberg, J.A. 1996b. Decision Model for the Optimization of the Plant Shutdown Criteria and Test Intervals. In: Probabilistic Safety Assessment PSA '96, September 21-October 3, 1996, Park City, Utah, USA. ANS, La Grange Park, USA, 1996. Pp. 355-362.

Holmberg, J. 1996c. A utility theoretic view on probabilistic safety criterion. To appear in research report series A, Helsinki University of Technology, Systems Analysis Laboratory, Espoo, Finland. 18 pp.

Holmberg, J., Hukki, K., Norros, L., Pulkkinen, U. & Pyy, P. 1996b. Contextual approaches to systems safety — analysis of decision making in an accident situation. In: P.C. Cacciabue & I.A. Papazoglou (eds.) Probabilistic Safety Assessment and Management '96— PSAM-III, Crete, June 24–28, 1996. London 1996, Springer. Pp. 1030–1035.

Holmberg, J. & Pulkkinen, U. 1995. Use of decision analytic methods in nuclear safety — an international survey. To be published in STUK report series. 21 p.

Lehtinen, E., 1995. Incorporation of the influences of organizational factors into PSA - a survey of the approaches and methodologies. VTT Automation, Industrial Automation, TAU-5005/95. 32 p.

NKS, 1996. Nordic Nuclear Safety Research, Summary report 1995 - Plans for 1996 and 1997. NKS(96)2, 1996-04-15.

NUREG-1150. 1989. Reactor Risk Reference Document. Washington D.C., U.S. Nuclear regulatory Commission, Office of Nuclear Regulatory Research.

Pulkkinen, U. 1994. Statistical models for expert judgement and wear prediction. Espoo: Technical Research Centre of Finland, VTT Publications 181. 65 p.+ app. 80 p.

Pulkkinen, U. & Pyy, P., 1996. A Bayesian Approach for Modeling Expert Judgements, In: P.C. Cacciabue & I.A. Papazoglou (ed.) Probabilistic Safety Assessment and Management '96— PSAM-III, Crete, June 24–28, 1996. London 1996, Springer.

Pulkkinen, U. & Virolainen, R. 1996. Use Of Expert Judgement In Finland. Report TAU-6021/96. VTT Automation, Industrial Automation. 5 p.

Pulkkinen, U. & Holmberg, J. 1996. A Method For Using Expert Judgement. Report TAU-6020/96. VTT Automation, Industrial Automation. 5 p.

Pyy, P. & Himanen, R. 1993. Shutdown risk analysis - lessons and views, SRA Europe, Fourth Conference, Rome, Italy, 18th - 20th October. 6 p.

Pyy, P., Reiman, L. & Laakso, K. 1997. Human errors in maintenance actions at nuclear power plants, STUK-YTO-Series XXX, under preparation.

Wahlström, B. 1996. Säkerhetsarbetet i Finland och Sverige, NKS/RAK-seminarium, 23 - 24 januari, Hasselbacken, Stockholm (In Swedish).

Wahlström, B. 1996. Systems safety in the high-tech industrial environments; technology and human reliability. Proceedings of the International Symposium Work in the Information Society, 20 - 22 May 1996, People and Work Research Reports 8, Finnish Institute of Occupational Health, Helsinki, Finland.

Wahlström, B. 1996. In control of nuclear safety. The Enlarged Halden Programme Group meeting at 19. - 23.5.1996, Loen, Norway.

Wahlström, B. 1996. Safety culture and communication; A comparison between Finnish and Swedish practices in nuclear safety, ESREDA Autumn Seminar / SRD Association Annual Conference - "Communicating Safety", 22 - 23 October 1996, Culham (Oxford), UK.

3.6 Human factors in NPP operations

The main objective of this study is to enhance safe operation of the nuclear power plant by providing knowledge of essential constraints and modes of the personnel's activity. A complex process can be controlled safely if at any time the process controllers are able to interpret correctly the state of the process on the basis of the available information. Therefore, on-line decision making in cooperatively organized process management is the core object of this study. The technical and social circumstances of the operators' performance are taken into account through the application of a contextual research methodology that has been developed during the study. Our approach reflects current international trends in the man-machine interaction research. The shared interest in contextual research approaches and systemic modeling concepts emerges from the needs for better control of risks in a dynamic society (Rasmussen 1996). The research is conducted in the *Human factors in NPP operations* project (ORINT).

We study an operators' activity in its dynamic interaction with its physical and social context. As the human is seen to act on the basis of the subjective significance of the activity, this has to be adopted as the methodological starting point for the explanation of his actions. As a consequence, our method can be characterized as subject centered. Moreover, we emphasize the dynamicity of activity. This refers to the notion that actions are situatively constructed, plans being resources of the organization of the course of action. Consequently, also a third methodological principle, contextuality can be identified, which emphasizes the necessity to analyze activity in its practical and social connections.

The decision making situation provides the context of the activity. Therefore it has to be described. We use a systemic modeling technique, with the help of which the situation specific boundary conditions of activity are described from the point of view of the available possibilities and constraints of performance. The performance will then be observed and registered with the help of various methods either in simulated or real-life situations. Analysis of the task performance is finally carried out with the help of the situation models, which can be simultaneously completed and modified. Through the analysis three major research topics are tackled, organization of task performance and learning, evaluation of the appropriateness of control room information, and evaluation of overall safety of the plant as a function of operators' performance.

Our research team has carried out parallel studies in other process control environments. This cross-domain approach has been found to be beneficial both for the development of the analysis method and for the practical applications.

3.6.1 Analysis of control room operators' on-line decision making

In this subproject the operators' cooperative decision making in complex process control situations was analyzed with the aim to enhance the comprehension of the critical decision-making demands of the operators' work. The central aim was to explain the dynamic organization of actions in natural work situations. We have based our assumptions concerning the dynamics of decision making on the Galperian concept of orientation (Galperin 1979). Orientation refers to the framing of the problem in a situation demanding action. Orientation indicates the logic according to which the task situation is taken into account by the actor. Therefore orientation has a major influence on the course of coping with the situation. (Norros 1989, Norros 1995, Norros 1996). As we have shown, orientations may be expressed by the subjects through the conceptions concerning the object of activity and the task situation (Norros 1995, Klemola & Norros in press). The subject's framing of the task situation can also be inferred on the basis of the subjects' habits of action, i.e. work practices and the conceptions regulating them. Habits of action can be specified through analysis of observations concerning the actors' utilization of the task and situation specific resources, and analysis of the explanatios he gives to his performance, in a real task performance (Hukki & Norros 1993, Hukki & Norros 1994). In the present study we analyzed the crews' task performance and habits of action in a disturbance situation with the aim, possibly to find evidence of the influence of habits of action on the efficiency of performance. The analysis method was developed during the analysis. In the following, we first present the experimental results (Norros & Hukki submitted) and then, in a separate section, the analysis method (Hukki & Norros submitted).

Dynamics of process operators' decision making in a disturbance situation. Results of a simulator study

A comprehensive simulator study was designed, which was carried out in cooperation with the TVO Olkiluoto NPP. It included a series of simulator experiments including a pilot experiment with two crews, and the main experiment with the remaining eleven crews of the TVO plant's two units. Each crew performed the same simulated disturbance situation as a part of their annual refresher training.

In the studied disturbance situation the task of the crew was to identify the state of the process after an evident process disturbance that caused a loss of feed water and the reactor level with the possible consequence of overheating the reactor (Fig. 64). The operators also had to choose methods to stabilize the process and prepare it for maintenance. The disturbance situation was first modeled from the decision-making point of view, according to the below described method, to provide reference for the analysis. The actual task performances of the crews were video-recorded and simulator logs including the operations and process events were collected during the simulator runs. Moreover, expert comments regarding the task performance and debriefing discussions with the crews were registered. The data was then analyzed in three-steps, description of the course of task performances, analysis of crews' habits of action, and the evaluation of the effect of habits of action on task performance.

The results are presented according to the phases of the analysis and a summary of the results is provided in Table 13.

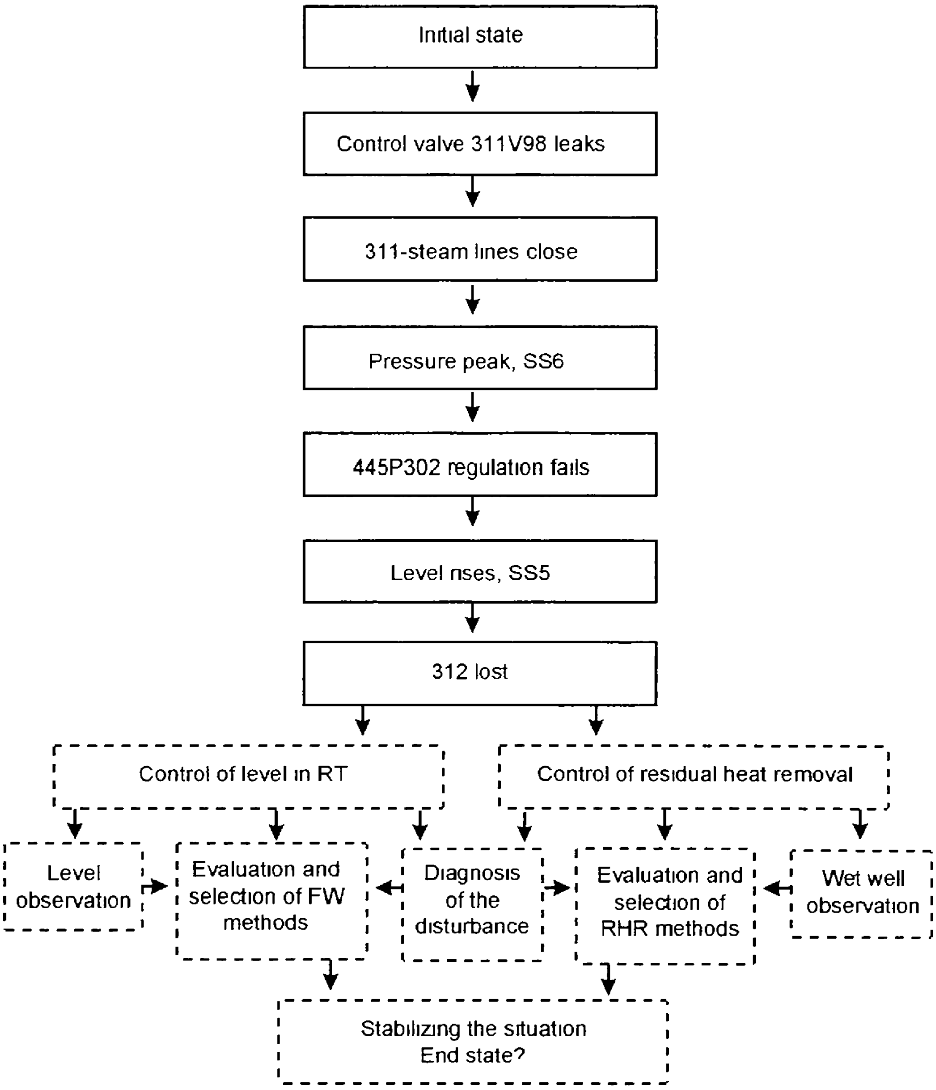


Figure 64. Functional description of the disturbance situation from a decision making point of view.

The crews' task performances were analyzed with the help of first defining the critical diagnostic actions (search for critical information, diagnostic inferences) choices of operating methods for maintenance of feed water and residual heat removal, utilization of procedures and contacting outside the control room. The time of performing each critical action was then indicated and courses of task performance were identified. Two performance patterns were identified (Table 13.). It was found out that crews differed in regard with integration of diagnostic activities in the stabilization of the process. One group of crews started to make explicit identification of the process state and inferences regarding the nature of the disturbance at a very early stage. Others prioritized operational actions in the initial phase of the disturbance. When the crews' further task performances were analyzed within both groups, it was found out that the crews differed also in regard with typical operational choices which reflected differences in taking account of the situational constraints. The crews' use of outside communication and Emergency Operating Procedures differed also (further details in Norros & Hukki submitted).

Table 13. Summary of the results of the 11 operator crews' decision making. The crews are indicated by capital letters.

Task performance	**Habits of action:** **-search for information** **-type of inferences** **-communication of inferences** **- attention to team situation**	**Effect habits of action on task performance**
Crews ABCIKL: Stabilization connected with identification of the disturbance situation resulting in situation specific operational choices. Communications outside were usually connected with own decisions. Use of EOP less accentuated.	Crews ABCI: interpretative habit of action	Crews ABCI: Effect on adequacy of process control was not evaluated. Provided high adaptability in novel situations through efficient utilization of available information and cooperative resources. Is supposed to promote learning from experience because situations are considered in their specificity.
	Crews LGH: way of acting could not be identified due to scarce communication and explication of inferential basis of actions.	
Crews DEFGH: Stabilization connected with the help of standard opera-tions. Communications outside were only partly connected with own decisions. Uniform use of EOP.	Crews DEFK: model-based habit of action	DEFK: Effects on adequacy of activity was not evaluated. Provided high efficiency of stabilization of the process in expected situations. Indications of vulnerability in novel situations, and for distractions. Also vulnerable for deficits in cooperation. Learning may cease because situations are interpreted to be routine.
		Crew K: the observed habit of action was not consistently connected with the task performance.

In the next phase, an analysis of the 11 crews' habits of action was carried out. This analysis was less comprehensive than the resulting method, because the structure and dimensions were only under development during the analysis. The second column of Table 13 provides the summary of the overall result of the analysis of these crews' habits of action. The results (presented in detail in Norros & Hukki submitted), particular patterns could be identified in the crews' habits of action. First, it became evident that process dynamic inferences seemed to correlate with high attention to process situation, which was interpreted to indicate an *interpretative habit of action* (crews ABCI). The disturbance model based inferences were related with a lower attention to the process situation, and can be characterized as a *model-based habit of action* (crews DEFK). Generally, effective utilization of team resources was typical to both of these patterns. Eight of the eleven crews could be identified to represent either of these two patterns. Beyond these major patterns, further characteristics of the way of acting could be identified. The crews FGH expressed lack of situativeness both in regard with the process and cooperative situation. The basis of inference remained undefined for crews GHL due to lack of communication of inference basis, but the crews GH seemed to manifest even a more general deficiency in the utilization of cooperative resources.

In the last step of analysis a comparison of the habit of action with the task performance of each crew was carried out. For practical reasons explicit evaluation of the adequacy of performance could not be carried out. The results of the analysis of the material allow, however, conclusions regarding the effects of the different habits of action on the adaptability of performance (see the last column of Table 13). First, according to our observations the interpretative habit of action which is based on process-dynamic inferences, provides higher adaptability in a particular situation, because the related task performance does not follow a predefined course of actions but is constructed according to the situation. Clear evidence of the efficiency of the interpretative habit of action was provided by two crews, who first made a false hypothesis about the nature of the disturbance. The crew that expressed an interpretative habit of action was able to correct the interpretation soon, whereas the crew expressing a model-based habit of action was stuck with the initial false hypothesis. The model-based habit of action relies on inferences based on a disturbance model, which, evidently, does not always fit into the situation. The model-based inferences are coupled with less active search for the process situation, a possible explanation of the well known tunnel-effect.

Situativeness, as an important prerequisite of adaptability, seems to have general relevance to the efficiency of habit of action. The above mentioned tunnel effect was related with lack of situativeness regarding both the process and the team situation. Beyond the tunnel effect, another typical phenomenon, distraction of activity due to external interruptions, has been brought up a severe problem in the control of complex environments (Weick 1993, Mandler 1982). In our material the task performances of two crews were distracted by interruptions of the normal expected flow of events. It can be argued that reorientation in the situation and reallocation of resources is less easy for those crews whose actions are organized according to the less situative habit of action regarding both the process and team situations.

The team provides an important resource for adaptive organization of performance. In their analysis of cooperative decision making in the cockpit Orasanu and Salas (1993) have demonstrated the significance of the pilot's communication of his inference basis for the efficiency of the crew's performance. According to our results, if combined with model-based inferences chief supervisors with deficient explication of the basis of inference and deficient use of other team members contributions, may be particularly prone to operating problems. It was further found that in case of operator crews with a long common work history, shared decision making may be coordinated even without communicating the basis of decision making, if the basis of inference was process-dynamic. Nevertheless, we think that the implicit nature of decision making is a source of uncertainty and potential problems in any unexpected situation. Problems may also arise if, in a disturbance situation, the staffing is incidentally not normal.

The results, allow even a more long-reaching conclusion. If, as is the case in the model-based habit of action, a process disturbance is basically interpreted as *a certain type of event,* known phenomena are expected to repeat and to be coped with routine. But, instead, if the process disturbance is considered as a *unique process phenomenon,* typical to the proponents of the interpretative habit of action, it spontaneously raises curiosity, search for information and contextual explanations of the observed events. As a consequence, new information of the process will be created, and learning from experience takes place. Therefore, it can be assumed, that beyond being more effective through contributing to adaptability of performance in a situation, the interpretative habit of action also promotes learning from experience. Verification of this assumption is one of the research topics in our future cooperation with the power plant.

The anchoring of the analysis of decision making in the concrete task situation with the help of the situation models is an essential feature of the method. While providing ecological validity to the analysis of decision making this solution also creates credibility of the analysis among the operators, because the basis of the ratings can be made explicit and discussed. This has, furthermore, enabled the design of the practical training tool.

The method for the analysis of the habit of action

The basic structure of the method for the analysis of habit of action (AHA) was developed during the study described above. This could be called an experimental phase in the development of the method. Two subsequent phases, a development and an implementation phase, were carried out in close cooperation with the TVO Olkiluoto power plant including two series of simulator experiments with different scenarios. A study that we carried out for the Barsebäck Nuclear Power Plant in Sweden, in which the availability of safety panels were evaluated through extensive simulator studies, also contributed to the development of the method as we were able to test our method in four more scenarios.

The developed method that was called analysis of habit of action (AHA) comprises of the following three phases of analysis (see details in Hukki and Norros submitted):

1. Choice and modeling of the task situation
2. Observation of the operators' performance
3. Evaluation of the dynamics of decision making

The *modeling of the situation* is carried out by describing the demands set by the process for appropriate carrying out of the control task. This means that a disturbance situation, or other dynamic situation, is conceptualized by defining the changes in the global state of the process, the critical indicators of the process state and control systems and the relevant control room operations. The task is divided into identification of the loss of stability of the process, stabilization of the process, and identification of the causes of the disturbance. Fig. 64 depicts a functional description of a disturbance situation from the decision making point of view used in our simulator studies. Possibilities and constraints of their successful performance. Different operational possibilities to control residual heat removal (RHR) are given with indication of their usability according to four criteria. Further models regarding available critical information in different decision situations etc. have also been provided regarding the disturbance. Finally the major problem solving demands are identified, and relevant procedures are also indicated.

The models, which can be made in the form of diagrams, tables etc., are conceptual tools which help to clarify how the operators utilize the available process information in the construction of their collective interpretation of the situation and on which basis they choose among the alternative operating methods in order to achieve their operational goals.

The created models are tentative and developing in the sense that the results of the subsequent analysis of the operators' activity and debriefing discussions with them may accomplish or even alter these definitions.

In the *observation of the operators' performance* during simulated or natural situations regular registration methods are used. Following data is supposed to be collected:

- expert observations of the crew's disturbance handling during the run with the help of prepared note sheets
- videotaped performance including operators' communications which are later transformed into written protocols
- trends of selected critical process parameters.

If researchers are supposed to make further analysis of the performance it would be necessary to get some additional registrations:

- simulator logs of process events
- simulator logs of operations carried out by the operators.

After the run a debriefing discussion is carried out during which the operators of the crew are asked to comment on their performance and give arguments concerning their decisions. The functional description of the task situation is used as a common tool in the discussion, in which the operators are encouraged to express their own interpretations of the situation.

On the basis of the collected data an *evaluation of the dynamics of decision making* will be carried out in three phases:

1. Description of task performance. The task performance refers to the observed sequence of process control actions and an evaluation of the adequacy of task performance with the help of defined process-based criteria. The time of identification of loss of stability, operational decisions, and identification of the causes of the disturbance, important communications outside the control room, cooperative actions within the crew, and the use of procedures are registered.

2. Analysis of habits of action. In this step, we then clarify the operators' habits of action through analysis of the interactions between the process and the operators from the point of view of the operators' utilization of available resources.

Habit of action expresses the way the subject organizes his activity in order to construct adaptive interactions with the environment. It includes the work practices and the conceptions regulating them and expresses the person's own accounts of the situation specific constraints and possibilities of the task.

The task, reflecting the intentional and cooperative nature of activity, is the first building block for the definition of the habit of action. Moreover, habit of action becomes manifests in the task performance as the *utilization of available resources,* which therefore has to be identified. In the NPP case the main resources were process information, the person's own resources, the team, and the procedures. Finally, we have suggested that, due to the subject's necessity to form an adaptive interaction with the environment his habits of action would have to express three *aspects of orientation*, coherency, situativeness, and reflectivity. The first aspect refers to the need to make coherent explanations of the world, the second to the necessity to account the variability and dynamic nature of the world, and the third to subject's less or greater ability to monitor the use of his own resources and efforts in coping with the environment. The developed method thus consisted of task-structured and resource-related items expressing the concrete criteria for evaluation of the habit of action. The three aspects of orientation form the theoretical background for the criteria that are used to distinguish between habits of action.

The current version of the method includes 40 items, out of which 16 express decision making concerning the process, 18 items relate to crew communication cooperation, and 6 items relate to coping with problem situations. Each evaluation is behaviorally anchored in the task with the help of the situation models, and in the observed task performance. Use of procedures is included in the method but the items and concrete evaluation criteria are not yet defined, because more information seemed necessary to find relevant criteria.

As a result ratings of operators' task performance in a specific task situation are achieved. These are further summarized into an evaluation of the crews' habits of action through ordering the rating results under the above main task demands and eleven dimensions of habits of action (the items under the fourth heading are descriptions not summaries of ratings, Table 14).

Table 14. Dimensions of habits of action ordered according to main task demands.

The crew's ways of decision making
- To what extent the crew has taken into account the global state of the process during the disturbance
- To what extent the crew has taken into account the particularities of the disturbance
- To what extent the crew has taken into account the situational possibilities and constraints of action
- To what extent the crew has taken into account the future demands of maintaining production

Team work
- To what extent the shift supervisor / the crew members have promoted to a shared interpretation of the situation
- To what extent the shift supervisor / the crew members have promoted to coherency of team cooperation

The crew's ways of coping with problem situations
- To what extent the shift supervisor / the crew members have been able to reorientate in the problem situation
- To what extent the shift supervisor / the crew members have been able to evaluate own resources critically

The crew's ways of using procedures
- Diagnostic use of procedures
- Use of procedures in stabilization of the process
- Use of procedures in coordination of team work

3. Evaluation of the effect of habits of action on task performance. This step of analysis consists of a comparison of the different dimensions of habit of action with the process control task, divided in several subtasks. Based on this, an overview of the dynamics of the crews' decision making can be achieved.

As a separate task we have developed a specific application of the above described method for the TVO NPP to be used as a regular training tool in simulator training. The direct benefits of the utilization of the method are enhancement of performance feedback for the trainees, and systematization of the training goals. The use of the method also provides possibilities for a follow-up of the development of operators' competencies. A new regulatory demand has recently come to force in Finland, according to which the power companies are obliged to organize performance-based competence tests in simulated disturbance sequences. These proofs complete the regular knowledge-oriented licensee tests. The AHA-method has been approved by the Finnish nuclear authorities as a means for the demonstration of the control room crews' competencies.

3.6.2 Development of the appropriateness of control room information

The central objective has been the development of an evaluation method for the validation of the appropriateness of control room information. By far the structure of the validation concept is available. The particular methodology used in the validation promotes understanding of how the available information resources provided by a certain interface are utilized. Based on this, judgments concerning the benefits of the interface can be made and understanding of operators' habits of action will be enhanced, which should have relevance for future training. During the validation process the operators' performance is studied from the point of view of how they as experts utilize available situational resources. The validation concept is currently used in a concrete validation task.

The essential questions in the validation are: does the system provide relevant information for the operators, and does it enhance the adequacy and safety of operators' performance in disturbance or other dynamic situations. The validation process consists of three phases: 1) analysis of information demands and supply, 2) analysis of the utilization of information during task performance and, 3) evaluation of the appropriateness of the information presentation.

In the analysis of information demands and supply the relevant information is firstly defined on the basis of an analysis of general information requirements. The suggested general requirements for information systems are the systems' informativeness, flexibility, reliability and ability to develop. Informativeness means the ability of the information system to provide possibilities to fulfill the demands set by the task by mediating relevant information. At least two kinds of informativeness can be identified: informativeness in regard with the functions and states of the process to be controlled (process informativeness) and informativeness in regard with the functions and states of the information system (system informativeness). Flexibility refers to the system's ability to provide possibilities to select information or change context according to situational demands and to operate in a way that it is not too restricted to preplanned situations. Reliability is a basic requirement of information system for fulfilling its mediating role because it affects the adequacy of activity and the credibility of the system. The ability of the system to develop refers to the possibility to complete or redesign the system after its implementation. It could also be interpreted as the system's potential to develop the activity of its users. Through taking account of the type and purpose of the system and the task situation, we create context dependent concrete criteria for information presentation. They indicate the way information should be presented and system functions designed in order to provide possibilities for adequate task performance.

The analysis of the utilization of information during task performance is analyzed with help of the AHA (analysis of habits of action) method.

In the third phase of the validation process the evaluation of the usability of the information presentation can be made. The evaluation of the concrete interface properties is based on operators' expert opinions and on the analysis of operators' task performance in real or simulated situations.

In 1997 the validation concept will be developed further. This means 1) development of a taxonomy of process control situations and 2) analysis of operators' conceptions of the usability of the available control room information.

The taxonomy is needed in order to gain a more comprehensive understanding of the situational demands of decision-making. The taxonomy is an essential part of the contextual approach under development at the man-machine research group. It is needed in the selection of situations for empirical studies, and it also contributes to the evaluation of the validity and generality of the results achieved in studies based on these particular situations. Furthermore, it is useful in the selection and design of scenarios for simulator training programs. The task involves conceptual work with the aim of describing critical features of process control situations. These features define the task demands that situations set for the operators, whose goal is an efficient and safe control of the process. The results of this task contribute to all ongoing research topics, i.e. control room design, development of operators' competencies and control of safety.

3.6.3 An integrated approach to system safety

As is widely agreed the current risk analysis of human performance in accident situations gives too static descriptions of the situation. It has also been pointed out that human contribution to risk has been considered in a too simple way in the current analyses. The goal of our study is to develop a more comprehensive methodology to risk analysis by integrating two different views, the probabilistic modeling based on decision analysis, and the contextual human factors psychology (Holmberg et al. 1996). The concrete aim of integration is to develop a conceptual interface between these views.

Definition of the tasks and some preliminary results

The realization of the above mentioned aims require particular prerequisites which have emerged at VTT Automation within the multidisciplinary research team. The team consists of psychologists and mathematically oriented risk analysts who share a common conception of the nature of decision making. While rejecting the deterministic model of decision making we attempt to find ways to conceive decision making from a more dynamic perspective. Regarding the probabilistic analysis this general attitude has been expressed in an attempt to complete PSA by describing the stochastic dynamics of the sequence including decision making. A potential modeling technique has even been suggested for this, i.e. the marked point process model (Arjas & Holmberg 1995, Holmberg 1997). A reorientation becomes necessary also regarding the psychological studies of decision making. In agreement with recently expressed concerns about the ecological validity of the traditional decision making paradigm (e.g. Cannon-Bowers et al. 1996), we would like to comprehend human activity as being in dynamic interaction with its physical and social context, reflecting decision makers' interpretations of the situation.

The course of our shared work to integrate the psychological and stochastic point of view in risk analysis has so far proceeded as follows:

1. Common description of the decision making context. We selected a particular case to be studied jointly. This was the cold over pressurization accident sequence. We then conceptualized the events in this sequence from the decision making point of view with help of causal description. Then, according to modeling technique introduced in the first section, a general description of decision making was prepared, completed with the definition of critical information for decision making and evaluation of usability of operational possibilities. These descriptions served as reference for modeling the decision making

2. Integrated modeling of the dynamics of decision making. In this phase we continued with the case study through carrying out a dynamic marked point process analysis. This was, furthermore, completed with an expert judgment model. Subsequently we carried out a simulator experiment at the TVO Olkiluoto training simulator and interviewed the crew.

As a results of the case study we could elaborate and verify the marked point process model on the basis of the simulator experiment. Moreover, we could state that from the safety point of view, the spurious start of an extra high head pump was a more important factor than decisions on pumping methods.

3. Integrated modeling of the dynamics of decision making. Regarding our goal of developing a more comprehensive risk analysis method we could state that the psychological analysis contributed to probabilistic modeling by helping to elaborate the marked point process model and improving the interpretation of the relevance of the information critical to the operators' decision making and risk. We also concluded that because simulator experiments simplify situations, other sources of information are needed to make a more realistic description.

In the future work we are proceeding to the next phase of integration. In this phase a new more comprehensive experimental data will be available. We will deepen the psychological analysis through the description of operators' ways of acting. According to our earlier studies (Norros & Hukki submitted) we can assume that the habits of action reflect operators' interpretations of the situation and, therefore also influence the course of their task performances. Based on this we will ask two questions: Is taking risk into account a distinct feature of operator's habits of action, and is there an effect of operators' habits of action on approaching the critical safety boundaries of the system?

References for section 3.6

Arjas, E. & Holmberg, J. 1995. Marked point process framework for living probabilistic safety assessment and risk follow-up. Reliability Engineering and System Safety 49, pp. 59 - 73.

Cannon-Bowers, J.A. Salas, E. & Pruitt, J. 1996. Establishing the boundaries of a paradigm for decision-making research. Human Factors, Volume 38, No 2, pp. 193 - 205.

Galperin, P.J. 1979. Introduction to psychology (in Finnish). Helsinki: Kansankulttuuri.

Holmberg, J. Hukki, K. Norros, L. Pulkkinen, U. & Pyy, P. 1996. An integrated approach to systems safety. Contextual analysis of decision making in an accident situation. Proceedings of the Probabilistic Safety Assessment and Management III Conference (PSAM III), June 24 - 28, Crete, Greece.

Holmberg, J. 1997. Probabilistic safety assessment and optimal control of hazardous technological systems. A marked point process approach. VTT Publications 305. Espoo: Technical Research Centre of Finland. 32 p. + app. 77 p.

Hukki, K. & Norros, L. 1993. Diagnostic orientation in control of disturbance situations. Ergonomics, 36, pp. 1317 - 1328.

Hukki, K. & Norros, L. 1994. A method for analysis of nuclear power plant operators' decision making in simulated disturbance situations. In: Norros, L. (ed). XIII European Annual Conference on Human Decision Making and Manual Control. VTT Symposium 146, Espoo: Technical Research Centre of Finland. Pp. 200 - 217.

Hukki, K. & Norros, L. (submitted). Analysis of operators' ways of acting (AWA). Development of a simulator training method. Le Travail Humain.

Klemola, U.-M. & Norros, L. (in press). Analysis of the anesthetist's activity: Recognition of uncertainty as a basis for new practice. Submitted to Medical Education.

Mandler, G. 1982. Stress and thought processes. In: Goldberger, L. & Breznitz, S. (eds.) Handbook of Stress. New York: Free Press. Pp. 88 - 164.

Norros, L. 1989. Responsibility for system development as an element of process operators' professional expertise. In: Proceedings of the 2nd European Meeting on Cognitive Science Approaches to Process Control. Siena, Italy.

Norros , L. 1995. An orientation-based approach to expertise. In: Hoc, J.-M. Cacciabue P.C. & Hollnagel, E. (eds.). Cognition and human computer cooperation. Hillsdale, NJ: Lawrence Erlbaum Associates. Pp. 137 - 160.

Norros, L. 1996. System disturbances as springboards for development of operators' expertise. In: Engeström, Y. & Middleton, D. (eds.). Cognition and communication at Work. Cambridge: Cambridge University Press. Pp. 159 - 175.

Norros, L. & Hukki, K. (submitted). Dynamics of process operators' decision making in a disturbance situation. A contextual analysis. International Journal of Cognitive Ergonomics.

Orasanu, J. & Salas, E. 1993. Team decision making in complex environments. In: Klein, G.A., Orasanu, J., Calderwood, R. & Zsambok, C.E. (Eds.). Decision making in action: Models and methods. Norwood, NJ: Ablex Publishing Corporation. Pp. 327 - 345.

Rasmussen, J. 1996. Risk management in a dynamic society: A modeling problem. Keynote address: Conference on Human Interaction with Complex Systems, Dayton, Ohio.

Weick, K.E. 1993. The vulnerable system: An analysis of the Tenerife air disaster. In: Roberts, K. (ed). New challenges to understanding organizations. New York: Mcmillan Publishing Company. Pp. 173 - 198.

4 Summary

4.1 LWR fuel performance

The objectives of the *Transient models of nuclear fuel* project (PATRA) were to acquire essential knowledge for assuring the safety of a high burnup fuel in transient conditions and to update the corresponding fuel performance models.

An agreement on co-operation in the field of nuclear safety was reached between the French IPSN (Institute de Protection et de Sûreté Nucléaire) and VTT. The agreement allows VTT to use the SCANAIR code. The applicability of the multidimensional mechanical program EPFMD developed at VTT has been improved. Besides thermoelastic strains and plastic deformations, the creep component has been added to time dependent applications. A test case showed encouraging results for long term cladding creep in comparison with measurement.

Good impression on ENIGMA steady state fuel performance code capabilities was confirmed by the international code benchmarking programme FUMEX organised by IAEA. As with most of the codes, the mechanical interaction could not be predicted well enough.

Due to the problems in the MR test reactor operation in Moscow and delays in finding an alternative experimental facility for the investigations, the Russian-Finnish *VVER fuel experiments* research programme (SOFIT) was terminated at the end of 1995. Part of the in-pile data and PIE data were released and submitted to NEA to be included in an international data base.

The Rod Over pressure Experiment ROPE II in Studsvik, Sweden demonstrated that a PWR fuel rod with an internal over pressure can be operated at about 200 W/cm for 3.5 months without measurable decrease in the gap conductance.

4.2 Reactor physics and dynamics

A comprehensive and independent reactor physics and dynamics code system has been created at VTT Energy for both BWR and VVER reactors. The code system is continuously modernised both by own development work and by international co-operation. At present most of the licensing analyses of the Finnish nuclear power plants can be performed with our own codes. The code system has been widely used by the nuclear safety authorities and by the Finnish nuclear power companies as well as by customers abroad.

The main objective of the *Calculation methods of reactor physics and dynamics* project (DYNAMIC) is to upgrade and extend the reactor physics and dynamics code system for tasks that are related to increased requirements and recent safety concerns. Applicability of the reactor physics codes needs to be ensured also for high burnup fuel. New methods have to be developed for the production of reliable and validated input data for reactor dynamics codes and other nuclear applications. The Monte Carlo technique is used for complex systems. Separate reactor physics codes are integrated into a validated calculation system. In reactor dynamics the objectives are to complement and validate the calculation system for complex reactivity accidents, such as ATWS, boron dilution and BWR core stability. The thermal hydraulic models of the dynamics codes are improved by taking into use the numerically accurate solution method PLIM and the physically based two-phase flow model SFAV.

A new code TRAB-3D has been developed for three-dimensional dynamic analyses of LWRs with rectangular core geometry. The rectangular solution method based on the nodal flux model of the HEXBU-3D and HEXTRAN has been successfully validated against accurate fine-mesh finite-difference calculations. Excellent agreement with POLCA4 results has been obtained in stationary full power calculations for a realistic core of the Finnish Olkiluoto BWRs. Calculations of three-dimensional NEACRP control rod ejection and control group withdrawal transients generally show a good agreement with the reference results of the benchmark. The core model will be connected with existing BWR and PWR circuit models at VTT, and the validation of the code will continue with comparisons against plant data of startup and transient measurements of the Olkiluoto reactors. The main applications of TRAB-3D will be transient and accident analyses (RIA, ATWS, stability, etc) for BWRs.

The new version of the one-dimensional TRAB-code utilizing VTT’s new hydraulics solution method in the core has been successfully tested in both VVER and BWR conditions. Results for the benchmark transients agree well with the results of the production version of TRAB. The new method totally eliminates numerical diffusion and dispersion, e.g. improving the tracking of boron and temperature fronts during transients. A simplified model for a VVER circuit utilizing the new solver has also been completed. As the next step the new solution method will be taken into use in the BWR circuit modelling.

The new six-equation model for two-phase flow based on the separation of flow according to velocity has been further developed and validated against measurements. The model has been successfully applied in BWR conditions to calculate response of the flow due to an oscillating inlet disturbance. The validation and testing will continue with applications to critical flows, counter-current flows, horizontal flows and flows with greatly varying void fraction. The long-term goal is to apply SFAV in VTT's reactor dynamics codes.

The reactor physics calculation system has been updated by taking into use many international codes, eg. the latest version of the NJOY nuclear data processing code, the SCALE4.3 program package, the MCNP4A Monte Carlo code, the advanced three-dimensional two-group reactor analysis code SIMULATE-3 of Studsvik Core Analysis Ab, and the first production version of Studsvik's fuel assembly program CASMO-4 (for square, hexagonal and cluster geometry). The NJOY and TRANSX processing codes have been applied to generate program-wise continuous energy and multi-group nuclear data libraries from the evaluated nuclear data libraries. Good agreement with reference results was achieved in validation calculations with different codes using the generated libraries.

VTT’s expertise in three-dimensional dynamics calculations has been utilised in defining, solving and coordinating three-dimensional hexagonal dynamics benchmarks in the international co-operation on VVER reactor physics and safety (AER). Three asymmetric control rod ejection benchmarks of increasing complexity are now followed by a benchmark on a reactivity accident initiated by a local boron dilution in the core. The next benchmark problem is aimed to include also the cooling circuit models.

The most important application area of the calculation system during the first two years of the DYNAMIC project has been the modernising and power upgrading projects of both Finnish nuclear power plants. The applications include the projects on feasibility and selection of modern fuel types, and demanding complicated transient and accident analyses.

4.3 Thermal-hydraulic experiments and analyses for VVER and ALWR plants

The objective of the *Thermal hydraulic experiments and analyses* project (TEKOJA) is to study thermal hydraulics of VVER-type PWRs in accidents and transients. The experiments belong to two categories. The first group includes experiments looking into the basic thermal hydraulics of VVERs. The objective of these experiments is to study influences of specific features of VVER reactors, such as the horizontal steam generators or hot and cold leg loop seals, on the natural circulation characteristics. The second group includes Accident Management (AM) experiments. The objective of these experiments is to study possible AM measures for VVERs. The third objective of the experiments is to provide data for validation of thermal hydraulic computer codes, such as APROS.

To fulfil the first objective, PACTEL operators ran a series of natural circulation experiments with different primary and secondary pressures. The experiments widened the data base on VVER natural circulation to low primary and secondary pressures. The experiments showed that pressure is an important parameter for natural circulation. Pressure influences the mass flow rate characteristics, and transition positions of different natural circulation modes. SBLOCA and primary secondary leakage (PRISE) experiments fulfilled the second objective. The experiments focused on the possibility of forming a non-borated water plug in the primary circuit. In SBLOCA's, forming of a non-borated water plug was possible, but the situation was very sensitive to the break size. In PRISE situations, flow reversal in the steam generator, i.e. flow of non-borated water from the secondary- to the primary-side, was clear in the experiments where the operators used primary feed and bleed.

In 1997 the main objective is to study some aspects of Anticipated Transients Without Scram (ATWS). These experiments start with two basic series. The first series, which was conducted in the first part of 1997, focused on start-up of two-phase natural circulation from the boiler condenser mode by utilising pressurizer water. The second series will investigate a maximum pressure scenario in control rod withdrawal. The ATWS experiments will continue in 1998 with a series simulating feed back from the core temperature and void fraction to the core power. The second series in 1998 will study the influences of non-condensable gas on natural circulation. This series will include experiments with different natural circulation modes and non-condensable gas concentrations. The third series will include an experiment with a steam generator collector rupture.

The first objective of the *Passive safety injection experiments* project PAHKO is to study a passive safety injection system behaviour in SBLOCAs. The investigated passive safety injection system consists of a passive safety injection tank (CMT), a cold leg pressure balancing line, and an injection line to the downcomer. The second objective is to simulate experiments with the APROS code. The first and second experiment series of PAHKO experiments have been completed and analysed. There were no problems with rapid condensation in the first series, such as observed in the earlier PACTEL experiments. In the second series, problems occurred when the operators removed the flow distributor (sparger) from the CMT. The analyses of the experiments supported application of McAdams correlation for calculation of heat transfer from the hot liquid layer to the CMT walls. The experiment analyses suggested the use of the Nusselt's film condensation correlation for calculation of condensation to the CMT walls. The main interest in the upcoming experiments is the effects of very small break accidents, where the recirculation phase of CMT operation is long. If the recirculation phase is long, the whole CMT may become full of hot water before the CMT starts to inject. This might cause problems for start-up of safety injection. The APROS code was used to simulate the first series GDE-24 experiment. The main simulation problem was the oscillation of the safety injection mass flow rate. The problems occurred because the code couldn't accurately simulate thermal stratification in the CMT.

The main objective for 1997 is to analyse the second experiment series, and to run the third series of PAHKO experiments. In the third series, the CMT will be moved to a higher position to increase driving head for safety injection. The series will also include studies of influences of operator actions on the CMT behaviour. The second objective of 1997 will be to continue analyses of the experiments with the APROS code. The main objective of 1998 will be to analyse and document the third series of PAHKO experiments, and to write the final report of the project. The second objective will be to analyse experiments from the third series with APROS code and to draw conclusions of the APROS code applicability for simulation of passive safety injection systems.

4.4 Severe accident management

One of the key objectives of the *Severe accident management* project (VAHTI) and its successor, the *Reactor accidents' phenomena and simulation* project (ROIMA), have been to validate thermal hydraulic models of the APROS code system. The APROS thermal hydraulic description has been extended to shutdown state conditions. The renewed code was tested by participating to the OECD international standard problem ISP 38, simulating the mid-loop operation of a PWR. A plant model of the Olkiluoto BWR has been created with the APROS code. The model has been tested by calculating three transients: steam line break with loss-of electricity, loss-of-feedwater and simultaneous closure of all steam isolation valves. The APROS code performed rather well in calculating the transients. Improvement needs were identified for some submodels.

The APROS code has been extensively applied in the thermal hydraulic license renewal analyses conducted for the Loviisa 1 and 2 plant units. The cases include large and small break LOCAs, steam generator primary to secondary leaks, loss of feedwater and loss of off-site power, closure of steam line isolation valves, pump trips and control rod withdrawal.

The RELAP5/MOD 3.2 code has been used to support the thermal hydraulic tests done with the PACTEL facility and as a reference for the APROS calculations. The ISP-38 transient was also calculated with the RELAP5/MOD 3.2 code. All major events of the test were predicted rather well by RELAP5, demonstrating that the code is suitable for use under atmospheric pressure conditions.

A computer code PASULA has been developed for analysis of core melt-pressure vessel interactions. The code has been applied to pressure vessel penetration analysis. The results showed that the nozzle construction is essential to determine the reactor pressure vessel failure mode. Probability of a large corium leakage through the control rod penetrations was considered low for the BWR type analyzed. On the other hand, instrument nozzles will fail quickly, and failure initiation at an instrument tube is much more probable than at a control rod. A creep model and possibilities to simulate porous debris have been recently added into the PASULA code system. Effective thermal conductivity estimated with the latter model has been compared with data found in the literature. The results agree well with most of the available data.

Cooling of a partially degraded core has been studied as a joint Nordic co-operation. The calculations were carried out for the Finnish Olkiluoto and Swedish Forsmark power plants with the MAAP4, MELCOR 1.8.3, and SCDAP/RELAP5 codes. The base scenario was a station blackout with variations in depressurization of the reactor coolant system. A set of restoration times was assumed for power and start of coolant injection. Rapid core cooling was obtained when reflooding was started at maximum core temperature of < 1600 K even with half of the total capacity of the high pressure injection system. The core was still coolable if the maximum cladding temperature at the beginning of reflooding was < 1800 K. All the codes predicted a different core end state after reflooding: MAAP4 predicted formation of a melt pool in the core, MELCOR resulted in formation of rubble bed and SCDAP/RELAP5 predicted material melting and/or fuel fragmentation due to mechanical stresses caused by temperature difference of coolant and hot fuel rods

The studies were extended to a case where reflooding was assumed after debris slump into the pressure vessel lower head. The analyses were conducted with the MELCOR code. It was found out, that reflooding could not terminate the accident progression after dryout of the lower head. Generally, reflooding accelerated the lower head failure because the exothermic metal oxidation overrides the cooling effect. High zirconium oxidation was estimated for all cases studied.

Recriticality of a BWR core has also been studied as a joint Nordic activity. The approach was to use severe accident codes to estimate probable control rod/fuel configurations and to use separate reactor physics codes to calculate the reactivity effects. Three codes were applied: RECRIT, APROS and SIMULATE-3K. The VAHTI project did part of the RECRIT development and applications as well as the APROS studies. The results show that reflooding of a partly control rod free core gives a power peak of a high amplitude, but with a short duration due to the Doppler feedback. The energy addition is small and contributes very little to heat-up of the fuel. With continued reflooding the fission power increases and tends to stabilise on a level that can be some tens per cent of the nominal power. The recriticality was calculated to take place very locally near the rewetting front. The results have hence a strong dependence on nodalisation.

The CONTAIN code has been tested against the natural circulation and helium mixing experiments done at the VICTORIA ice condenser containment test facility. A characteristic feature of the experiments was a global natural circulation loop, which was relatively well estimated by CONTAIN. The facility pressure and temperature were slightly overpredicted at the late phase of the experiment.

Aerosol experiments have been conducted with the AHMED test vessel. The aim of the AHMED test program was to improve understanding of containment aerosol behaviour at known thermal-hydraulic conditions. The experiments were successful. Homogeneous temperature and humidity were achieved within the vessel. A total of 17 aerosol test were performed using NaOH, CsOH, CsI and Ag aerosol species. AHMED was the first facility in which the aerosol mass concentration could be monitored on-line and the wet aerosol size distribution was directly measured. The results filled a gap that existed on knowledge of hygroscopic aerosol behaviour. The data have been used by developers of aerosol models for severe accident codes. A CSNI calculation exercise has been arranged based on the AHMED tests.

A training tool for severe accidents is being developed by implementing selected severe accident models into the APROS code. The VAHTI-project has contributed improvements in the APROS containment model to describe the physical conditions and accident management hardware relevant for severe accidents. The phenomena currently modeled are steam/air/non-condensable gas mixture thermodynamics, condensation and evaporation on the heat structures and water droplets, heat transfer to and heat conduction in the heat structures, and intercell flows. The engineered safety features include the internal and external spray systems, ice condensers, suppression pool system, and hydrogen control devices. New models for the current code version comprise of non-condensable gas behaviour (others than air), its effects on mass transfer phenomena, and buoyancy effect.

4.5 Risk and reliability analyses

Use of probabilistic safety analysis (PSA) has become an important safety management tool. PSA, however, gives only an indicator of severe core damage risk. Thus, there are other criteria that have to be taken into account in safety related decisions, such as economy, maintainability and occupational safety. Neglecting them and looking only at PSA results may lead to bad decisions and even deteriorate safety culture.

Uses of multi-attribute decision analysis approach have been demonstrated to aid decision makers by giving a structured guidance on alternatives and their values. In future, an increased use of explicit decision analysis techniques is foreseen, and their development and uses will be promoted in the *Reliability and risk analyses* project (LURI). Uncertainty analysis and expert judgement are an important part of PSA work. Actually, they are used in every PSA, but mostly embedded in the models and data. In the LURI project, methods have been developed to express model uncertainty explicitly and to combine expert judgement with other evidence in Bayesian manner. Under the research theme 'risk based regulation', coupled closely to LURI, these issues will be integrated under the decision analysis framework to build a consistent probabilistic safety management approach.

Human reliability analysis has always been a problem area of PSA. On one hand, human behaviour does not necessarily obey the same laws as a component; on the other hand, the exclusion of human factor from PSAs would surely lead to a biased view on safety level. The LURI project has approached the problem by establishing co-operation with other scientific disciplines, e.g. psychology, and by using real NPP maintenance data to study human error frequencies. The preliminary conclusion of this work is that decision analytic and dynamic stochastic process models suit well to detailed modelling of human reliability. Similarly, human error data may be studied by using the same principles as component failure data. In the project the scope of human actions is also expanded to safety management and NPP modification practices, which seem to be important from actual safety management point of view but seldom present in PSAs.

4.6 Human factors in NPP operations

On-line decision making in co-operatively organized process management is the core object of the *Human factors in NPP operations* project (ORINT). In the work the technical and social circumstances of the operators' performance are taken into account through the application of a contextual research methodology that has been developed during this study. The work has been carried out in four parallel research topics in which a common methodology has been utilised:

1. Analysis of control room operators' on-line decision making. The operators' co-operative decision making in complex process control situations was analysed with the help of a full-scope process simulator. Comprehensive data was acquired of the operators' performance and the process state in a disturbance situation. The data was then analysed in three steps, description of the course of task performances, analysis of crews' habits of action, and the evaluation of the effect of habits of action on task performance. The analysis of the crews' courses of task performance revealed two major habits of action, interpretative and model-based ones, which seem to imply more or less adaptive control of the disturbance, respectively.

2. Method for the analysis of the habit of action. The basic structure of the analysis of habit of action (AHA) method was developed during the study described above. In addition, two subsequent phases were carried out in close co-operation with TVO utility. A study carried out for a Swedish NPP also contributed to the development of the method. The method comprises of three major phases: choice and modelling of the task situation, observation of the operators' performance, and evaluation of the dynamics of decision making. The central idea of the method is to define, with a rating tool, the operators' habits of action which are supposed to regulate the operators' ways of organising their task performances in dynamic process control situations. The efficiency of different habits of action can be evaluated by comparing habits of action with adequacy of task performance. The latter is defined according to selected criteria indicating the maintenance of the safe state of the process on a global level. As a separate task, a specific application of the method was prepared for the Finnish Olkiluoto NPP to be used as a regular training tool in simulator training.

3. Development of the appropriateness of control room information. The objective has been the development of an evaluation method for the validation of the appropriateness of control room information. Currently the structure of the validation concept is available, and the method has been used in one application. The results of the study are under analysis.

4. An integrated approach to system safety. The goal of this study is to develop a more comprehensive methodology to risk analysis by integrating two different views, the probabilistic modelling based on decision analysis, and the contextual human factors psychology. The concrete aim of integration is to develop a conceptual interface between these views.

In the future work two challenges are seen: The first is to develop the research methods into practical tools in order to serve daily needs of the plants. This requires close co-operation with the plant personnel, of which good experience already exists. In connection with the first challenge, the theoretical basis of the methodology will be elaborated. Most important tasks are: development of the technique for modelling of task situations, exploring effects of habits of action on learning, and deepening the analysis of decision making by better consideration of operators' ways of coping with problem situations.

ANNEX A

International co-operation

In the following catalogue the most important current international cooperation contacts for the research programme are listed together with the name and affiliation of the relevant contact person(s). (There are additional members in the various groups from utilities, STUK and KTM.)

OECD/Nuclear Energy Agency

Committee on the Safety of Nuclear Installations (CSNI), L. Mattila, VTT Energy

* *Principal Working Group No. 1 (PWG1): Operating Experience and Human Factors, and its Expanded Task Force on Human Factors,* K. Laakso, P. Pyy, VTT Automation
* *Principal Working Group 2 (PWG2): Primary Coolant System Behaviour,* H. Holmström, VTT Energy
    * *Task Group on Thermal Hydraulic Analyses (PWG2/TG-THA),* V. Yrjölä, H. Purhonen, VTT Energy
    * *Degraded Core Cooling Task Group (PWG2/DCC),* E. Pekkarinen, VTT Energy
* *Principal Working Group 4 (PWG4): Confinement of Accidental Radioactive Releases,* R. Sairanen, VTT Energy
    * *Containment Aspect of Severe Accident Management (PWG4/CAM),* H. Sjövall, TVO
    * *Severe Accident Phenomena in Containment Task Group (PWG4/SAC),* R. Sairanen, VTT Energy
    * *Fission Product Phenomena in the Primary Circuit and the Containment Task Group (PWG4/FPC),* J. Jokiniemi, VTT Energy
* *Principal Working Group No. 5 (PWG5): Risk Assessment*, U. Pulkkinen, VTT Automation
* *Verification Matrix for Thermal-Hydraulic System Codes Applied for WWER Analysis*, H. Holmström, VTT Energy
* *Support Group on PSB-VVER Project,* J. Kouhia, VTT Energy

Nuclear Science Committee (NEANSC), M. Anttila, VTT Energy

*	*Task force on scientific issues in fuel behaviour,* S. Kelppe, VTT Energy

*	*Chemistry task force,* U. Vuorinen, VTT Chemical Technology

Nuclear Development Committee (NEANDC), M. Kara, VTT Energy

Halden Reactor Project/Halden Programme Group, B. Wahlström, VTT Automation, L. Mattila, VTT Energy

*	*Fuel performance analysis,* K. Ranta-Puska, VTT Energy

*	*Concepts for control room solutions with regard to low power and shutdown states,* P.Pyy, B. Wahlström, VTT Automation

*	*Integrated human reliability modelling and simulator experiments (experiments financed partly by U.S. NRC),* P. Pyy, L.Norros, VTT Automation

*	*Reliability of software based control systems,* U. Pulkkinen, P. Haapanen, VTT Automation

*	*Advanced test bench for man-machine experiments,* K. Juslin, VTT Automation, O. Tiihonen, VTT Energy

International Atomic Energy Agency

Co-ordinated Research Projects

*	*Long-term storage of spent fuel (BEFAST III),* E. Vitikainen, VTT Manufacturing Technology

*	*Fuel modelling at extended burnup (FUMEX),* S. Kelppe, VTT Energy

*	*Acquisition and Utilization of Knowledge within Expert Systems,* H. Välisuo, VTT Automation

*	*Human reliability data collection,* P.Pyy, K. Laakso, VTT Automation

International Working Group on Water Reactor Performance and Technology (IWGFPT), R. Teräsvirta, IVO Power Engineering, S. Kelppe, VTT Energy

International Working Group on Nuclear Power Plant Control and Instrumentation (NPPCI), B. Wahlström, VTT Automation

Development of plant-specific safety indicators, E. Lehtinen, VTT Automation

Commission of the European Communities

Programme Committee for the Research Programme on Nuclear Fission Safety (NFS-2), L. Mattila, VTT Energy

* *Assessment of Passive Safety Injection Systems of Advanced Light Water Reactors*, Project Leader: J. Tuunanen, VTT Energy
* *European BWR-R&D Cluster for Innovative Passive Safety Systems*, H. Holmström, VTT Energy
* *In Vessel Core Degradation and Coolability (MVI),* K. Kilpi, VTT Energy, O. Kymäläinen, IVO Power Engineering
* *Revaporization,* J. Jokiniemi, VTT Energy
* *Aerosol Physics in Containment,* J. Jokiniemi, VTT Energy, H. Tuomisto, IVO Power Engineering
* *Concerted Action on Safety Related Innovative Nuclear Reactor Technology Elements - R&D Network*, R. Sairanen, VTT Energy
* *Fission Product Vapour/Aerosol Chemistry in the Primary Circuit (PC CHEM),* J. Jokiniemi, VTT Energy
* *Severe Accident Recriticality Analyses*, E. K. Puska, VTT Energy

Benchmark Excercise on Expert Judgment, U. Pulkkinen, VTT Automation

Concerted Action on Integrated Sequence Analysis, P.Pyy

Cooperation on VVER Reactor Physics and Dynamics (AER), H. Räty, VTT Energy

* *Working group on VVER Reactor Safety Analysis*, R. Kyrki-Rajamäki, VTT Energy
* *Working group on physical problems of spent fuel, radwaste and decommissioning of nuclear power plants,* M. Anttila, VTT Energy
* *AER annual symposiums,* H. Räty, VTT Energy

Nordic Nuclear Safety Research (NKS) 1994-1997, Steering group, L. Mattila, VTT Energy

* *Strategy for reactor safety, RAK-1,* P. Pyy, B. Wahlström, VTT Automation
* *Controlling Accident Releases RAK-2,* Project leader: I. Lindholm, VTT Energy

Cooperation with various institutes

Kurchatov Institute, Moscow Russia

* *The Finnish-Russian SOFIT Programme on VVER fuel behaviour*, R. Teräsvirta, IVO Power Engineering
* *Scientific cooperation between Kurchatov Institute and VTT Energy on thermal hydraulic experiments,* J. Kouhia, VTT Energy

Research and Engineering Centre of Nuclear Power Plants Safety, Electrogorsk, Russia

* *Scientific cooperation between EREC and VTT Energy on thermal hydraulic experiments,* J. Kouhia, VTT Energy

Studsvik, Sweden

* *Rod Over-pressure Experiments (ROPE II), Project Group,* J.-O. Stengård, VTT Energy
* *Defected Fuel Experiments (DEFEX), Project Group,* S. Kelppe, VTT Energy

Gesellschaft für Anlagen und Reaktorsicherheit mbH (GRS), Germany

* *UPTF/TRAM Programme,* J. Tuunanen, VTT Energy

Commissariat a l'Energie Atomique / Centre d'Etudes Nucleaires de Grenoble (CEA/CENG), Grenoble, France

* *Cooperation between CENG, VTT and LTKK on use of CATHARE code,* H. Kalli, Lappeenranta University of Technology

Institute de Protection et de Surete Nucleaire, Cadarache, France

* *SESAME software system and its adaptation to VVER-type NPP,* S. Vuori, VTT Energy
* *Study of the behaviour of highly irradiated fuels in case of reactivity accident and the SCANAIR computer code,* S. Kelppe, VTT Energy
* *Severe accidents software cooperation (ICARE),* L. Mattila, VTT Energy

Electric Power Research Institute (EPRI)

* *Advanced Containment Experiments, Extension (ACEX),* I. Lindholm, VTT Energy

* *Melt Attack and Coolability (MACE),* I. Lindholm, VTT Energy

US Nuclear Regulatory Commission (USNRC)

* *Co-operative Severe Accident Research Programme (CSARP),* R. Sairanen, VTT Energy

* *Code Application and Maintenance (CAMP),* H. Holmström, VTT Energy

European Safety, Reliability and Data Association (ESReDA)

* *General Secretary, Organisation of ESReDA Seminars,* P. Pyy, VTT Automation

Swedish Nuclear Power Inspectorate (SKI), Sydkraft and Vattenfall Ab, Sweden

* *Statistical methods, decision analysis, human errors, maintenance and PSA*, K. Laakso, P. Pyy, VTT Automation

International Institute for Applied Systems Analysis (IIASA)

* *Decision making and risk based regulation,* U. Pulkkinen, VTT Automation

Other co-operation

Nordic Reactor Physics Meetings "Reactor Physics Calculations in the Nordic Countries", R. Höglund, VTT Energy

International Seminars on Horizontal Steam Generator Modelling, H. Purhonen, VTT Energy

European Safety and Reliability Conferences (ESREL), P. Pyy, VTT Automation

ANNEX B

Publications in the Projects of the Reactor Safety Research Programme (RETU) in 1995 - May 1997

(Some reports in Finnish only)

Table of publications in the research fields or projects of the RETU programme during January 1995 - May 1997.

Research field or project	Acronyms of the projects	Ref. to section of this report	Scientific journals	Conference papers	Research institute reports	Others	Total
LWR fuel performance	PATRA	3.1		4		7	11
Reactor physics & dynamics	DYNAMIC	3.2	2	43	2	44	91
Thermal-hydraulic experiments and analyses for VVER and ALWR plants	TEKOJA and PAHKO	3.3		14		32	46
Severe accident management	VAHTI and ROIMA	3.4	1	20	4	27	52
Risk and reliability analyses	LURI	3.5	3	16	9	7	35
Human factors in NPP operations	ORINT	3.6	2	5	1		8
	Total		8	102	16	117	243

Transient models of nuclear fuel (PATRA) and VVER-fuel experiments (SOFIT)

Conference papers

Kelppe, S., Roine, T. & Lunabba, R. 1995. ENIGMA calculations on fuel irradiated in TVO I reactor compared with pool-side fission gas release measurements. Paper presented at the Topfuel ‘95 Conference on 12 to 15 March 1995 in Würzburg Germany. 4 p.

Ranta-Puska, K. 1995. Thermal effects of fuel thermal conductivity degradation and rim in IFA-597.2 (BWR) and IFA-533.2 (HBWR) rods. Paper presented in the Enlarged Halden Programme Group Meeting, Loen Norway 19 - 24 May 1996. 8 p. + fig.

Ranta-Puska, K.1996. IFA-503.1 (VVER/PWR Test): ENIGMA calculations compared with the first measurement data. Paper presented in the Enlarged Halden Programme Group Meeting, Loen Norway 19 - 24 May 1996. 5 p. + fig.

Pihlatie, M., Ranta-Puska, K.1995. Probabilistic Analysis of Loviisa Nuclear Fuel Rod Behaviour. Paper presented in the Second International Seminar on WWER Fuel Performance, Modelling and Experimental Support, Sandanski, Bulgaria 21 - 25 April 1997

Others

Stengård, J-O.1996. Summary of Studsvik ROPE II project. Espoo: VTT Energy. PATRA-8/95. 21 p. (in Finnish.)

Ranta-Puska, K.1995. Thermal effects of fuel thermal conductivity degradation and rim in IFA-597.2 (BWR) and IFA-533.2 (HBWR) rods. Espoo: VTT Energy. PATRA-9/95. 8 p.

Ranta-Puska, K. 1995. IFA-503 (VVER/PWR TEST): The pre-calculations against first irradiation data. Espoo: VTT Energy.Work report, 25.10.1995.

Ikonen, K. 1996.Creep in Thermo-Plastic Deformation and Stress Analysis. Espoo: VTT Energy. PATRA-3/96.

Ranta-Puska, K. 1996. Proposals of tasks on fuel behaviour research offered by VTT Energy for co-operation with the Halden Project in. Espoo: VTT Energy. PATRA-4/96, 21.11.96.

Kelppe, S.1996. Summary of Studsviks DEFEX project. Espoo:VTT Eneregy. PATRA-5/96. (in Finnish.)

Kelppe, S.1997. Progress of development, validation and application of the SCANAIR code. Espoo: VTT Energy. PATRA-6/96.

Reactor Physics and Dynamic Methods Including Thermal Hydraulics (DYNAMIC)

Publications in scientific journals

Narumo, T. & Rajamäki. M. 1995. Comparing the SFAV-based two-fluid model with virtual-mass models. International Journal of Power and Energy Systems, vol. 15, nro 1, pp. 29 - 36.

Rajamäki, M. & Narumo, T. 1995. A six equation SFAV-model for two-phase flow with correct propagating velocities of disturbances. Journal of Numerical Heat Transfer. Part B, vol. 28, nro. 4, pp. 415 - 436.

Conference papers

Antila, M., Siltanen, P, Kyrki-Rajamäki, R. & Vanttola, T.1995. Analysis of core response to the injection of diluted slugs for the Loviisa VVER-440 reactor. Proceedings of the OECD/NEA/CSNI Specialist Meeting on Boron Dilution Reactivity Transients. State College, PA 18 - 20 October 1995.

Gács, A., Keresztúri, A., Telbisz, M., Kyrki-Rajamäki, R. & Siltanen, P.1995. New safety analyses of RIA and ATWS events for Paks NPP. Proceedings of the International ENS TOPical Meeting TOPFORM '95, Today's Cost Competitive NPP for Current and Future Safe Operation, Avignon March 24 - 28 March 1995. Paris: Société Française d'Énergie Nucléaire. Pp. 319 - 331.

Gromov, A., Kalugin, S., Podshibyakin, A., Siltanen, P. & Kyrki-Rajamäki, R. 1996. Investigation of accidents with decrease of boric acid concentration in the primary coolant of VVER-1000/model 91 reactor plant. 1996. Proceedings of the 1996 ASME/JSME International ICONE-IV, The 4th Annual Conference on Nuclear Engineering, New Orleans, LA 10 - 14 March 1996. American Society of Mechanical Engineering. Pp. 127 - 139. (I0389C-1996.)

Höglund, R. 1997. BWR reloading using CORFU. 8th Nordic Reactor Physics Meeting "Reactor physics calculations in the Nordic countries". Kjeller. 13 - 14 March 1997.

Höglund, R.1995. The CORFU program for BWR core reload design. In: "In-core fuel management: reloading techniques". Proceedings of a Technical Committee Meeting and Workshop held in Vienna 19 - 21 October 1992. Vienna: International Atomic Energy Agency. Pp.85 - 92. (IAEA-TECDOC-816.)

Höglund, R., Roine, T. & Solala, M. 1995. Experience in the utilization of BWR fuel including core follow calculation with the CASMO-SIMULATE code package and efficient core design with the CORFU program. Transactions of the International KTG/ENS TOPical Meeting on Nuclear Fuel TOPFUEL '95. Würzburg 12 - 15 March 1995. Volume II Poster Papers. Pp. 161 - 164.

Höglund, R. & Solala, M. 1997. ICFM for a BWR using the CORFU core designer. Proceedings of the Topical Meeting Advances in Nuclear Fuel Management II, Myrtle Beach, SC 23 - 26 March 1997. La Grange Park, IL: American Nuclear Society. Vol 2. Pp. 17-13 - 17-20. (EPRI TR-107728-V1.)

Kaloinen, E. 1995. Development and validation of stationary physics code HEXBU. STUK-IAEA seminar "Transient and accident analyses and codes validation for WWER-440 reactors", Helsinki 13 - 17 November 1995.

Kaloinen, E.1997. New solution method for two-group diffusion equations in the dynamic code TRAB-3D. 8th Nordic Reactor Physics Meeting "Reactor physics calculations in the Nordic countries, Kjeller, Norway. 13 - 14 March 1997.

Kaloinen, E. & Kyrki-Rajamäki, R. 1997. TRAB-3D, a new code for three-dimensional reactor dynamics. ASME/SFEN/JSME ICONE-V, 5th International Conference on Nuclear Engineering. "Nuclear Advances through Global Cooperation", Nice, France 26 - 30 May 1997.

Kelppe, S., Roine, T. & Lunabba, R. 1995. ENIGMA calculations on fuel irradiated in TVO I reactor compared with pool-side fission gas release measurements. Transactions of the International KTG/ENS TOPical Meeting on Nuclear Fuel TOPFUEL '95, Würzburg. 12 - 15 March 1995. Volume II Poster Papers. Pp. 109 - 112.

Knott, D., Edenius, M., Peltonen, J. & Anttila, M. 1997. Results of modelling hexagonal and circular cluster fuel assembly designs using CASMO-4. Proceedings of the Topical Meeting Advances in Nuclear Fuel Management II, Myrtle Beach, SC 23 - 26 March 1997. La Grange Park: IL: American Nuclear Society. Vol 1. Pp. 5-1 - 5-15. (EPRI TR-107728-V1.)

Kyrki-Rajamäki, R. 1996. Application of coupled 3D neutronics and circuit thermal hydraulics in reactor analysis. Grenoble, France 2 - 3 December 1996. French- Finnish Colloquium on PWR Safety. (invited paper.)

Kyrki-Rajamäki, R.1996. Application of the best estimate dynamics code HEXTRAN to accident analyses. Proceedings of the 6th symposium of the AER on VVER Reactor Physics and Reactor Safety, Kirkkonummi, Finland 23 - 26 September 1996. Budapest: KFKI/AEKI. Pp. 51 - 57.

Kyrki-Rajamäki, R. Definition of the 4th dynamic AER benchmark. Proceedings of the 6th symposium of the AER on VVER Reactor Physics and Reactor Safety. Kirkkonummi, Finland 23 - 26 September 1996. Budapest: KFKI/AEKI. Pp. 237 - 246.

Kyrki-Rajamäki, R 1995. Description of VTT's code system for analyses of reactor systems. STUK-IAEA seminar "Transient and accident analyses and codes validation for WWER-440 reactors". Helsinki 13 - 17 November 1995.

Kyrki-Rajamäki, R.1996. On the role of burnup effects of fuel properties in RIA analyses. Proceedings of the Specialist Meeting on Transient Behaviour of High Burnup Fuel

(NEA/CSNI and CEA/IPSN), Cadarache, France 12 - 14 September 1995. Paris: OECD Pp. 269 - 278. (NEA/CSNI/R(95)22.

Kyrki-Rajamäki, R.1995. Reactor dynamics analyses for Loviisa NPP. STUK-IAEA seminar "Transient and accident analyses and codes validation for WWER-440 reactors", Helsinki 13 - 17 November 1995.

Kyrki-Rajamäki, R.1995. Reactor dynamics codes of VTT: HEXTRAN, SMATRA and the new hydraulic solver PLIM. STUK-IAEA seminar "Transient and accident analyses and codes validation for WWER-440 reactors", Helsinki 13 - 17 November 1995.

Kyrki-Rajamäki, R.1997. Recent development of the TRAB code family. 8th Nordic Reactor Physics Meeting "Reactor physics calculations in the Nordic countries", Kjeller, Norway 13 - 14 March 1997.

Kyrki-Rajamäki, R.1995. VTT Energy's program system for reactor physics and dynamics calculations. In: Proceedings of the 7th Nordic Reactor Physics Meeting "Reactor physics calculations in the Nordic countries", Espoo 8 - 9 May 1995. Espoo: Technical Research Centre of Finland. Pp. 67 - 78. (VTT Symposium 159.)

Kyrki-Rajamäki, R., Grundmann, U. & Keresztúri, A. 1996. Results of three-dimensional hexagonal dynamic benchmark problems for VVER type reactors. Proceedings of the PHYSOR '96, International Conference on the Physics of Reactors -Breakthrough of Nuclear Energy by Reactor Physics, Mito, Ibaraki Japan 16 - 20 September 1996. Mito, Ibaraki: Japan Atomic Energy Research Institute. Vol. 3. Pp. J-229 - J-238.

Kyrki-Rajamäki, R. & Kaloinen, E. 1995. Results of the third three-dimensional hexagonal dynamic AER benchmark problem including thermal hydraulics calculations in the core and a hot channel. Proceedings of the 5th symposium of the AER (Atomic Energy Research for Investigating Neutron Physics and Thermohydraulics Problems of Reactor Safety), Dobogókö, Hungary 15 - 19 October 1995. Budapest: KFKI Atomic Energy Research Institute. Pp. 255 - 286.

Kyrki-Rajamäki, R. & Stenius, T. 1995. Calculation of local boron dilution accidents with HEXTRAN code. Proceedings of the ANS Topical Meeting: International Conference on Mathematics and Computations, Reactor Physics, and Environmental Analyses, Portland, OR 30 April - May 4 1995. La Grange Park, IL: American Nuclear Society. Vol. 1. Pp. 274 - 283.

Narumo, T.1995. Improvements to accuracy of transient reactor power calculations achieved with PLIM (Piecewise Linear Interpolation Method) in hydraulics. In: Proceedings of the 7th Nordic Reactor Physics Meeting "Reactor physics calculations in the Nordic countries", Espoo, Finland 8 - 9 May 1995. Espoo: Technical Research Centre of Finland. Pp. 279 - 289. (VTT Symposium 159.)

Narumo, T. & Rajamäki, M.1996. Dividing phases in two-phase flow and modelling of interfacial drag. OECD/CSNI Workshop on Transient Thermal Hydraulic & Neutronic Codes Requirements, Annapolis, MD 5 - 8 November 1996. (invited paper)

Narumo, T. & Rajamäki, M.1996. Wall and interfacial friction terms of a dynamic whole-range two-fluid model. Transactions of the ANS/ENS 1996 International Conference, Washington, DC 10 - 15 November 1996. La Grange Park, IL: American Nuclear Society. Pp. 390 - 392. (TANSAO 75 1-490.)

Peltonen, J. 1996. High and second order polynomial feedback models for VVER-440 fuel two group cross sections. Proceedings of the 6th symposium of the AER on VVER Reactor Physics and Reactor Safety, Kirkkonummi, Finland 23 - 26 September 1996. Budapest: KFKI/AEKI. Pp. 333 - 343.

Puska, E.K., Norrman, S., Kyrki-Rajamäki, R., Siltanen, P., Porkholm, K., Latyeva, L, Semishin, V. & Kalugin, S. 1997. Advanced process simulation with APROS for VVER-91, the next generation reactor design for VVER-1000. 2nd International Conference on Advanced Reactor Safety (ARS'97), Orlando, FL 1 - 4 June 1997.

Rajamäki, M. 1996. Elimination of numerical diffusion in one-phase and two-phase flows. OECD/CSNI Workshop on Transient Thermal Hydraulic & Neutronic Codes Requirements, Annapolis. MD 5 - 8 November 1996. (invited paper)

Rajamäki, M. 1996. Modelling of appearance, movement and disappearance of water-level-like discontinuity in channel geometry. European two-phase flow group meeting, Grenoble, France 3-5 June 1996.

Rajamäki, M. 1996. New methods in hydraulics. Grenoble, France 2 - 3 December 1996 French- Finnish Colloquium on PWR Safety. (invited paper)

Rajamäki, M. & Kyrki-Rajamäki, R.1995. Boron dilution transient calculation with the HEXTRAN code using CFDPLIM. Proceedings of the OECD/NEA/CSNI Specialist Meeting on Boron Dilution Reactivity Transients, State College, PA 18 - 20 October 1995.

Roine, T. & Anttila, M. 1995. Application of CASMO-4 in criticality safety studies at VTT Energy. CASMO Users' Group Meeting Europe, Hamburg 5 - 6 October 1995 .

Roine, T. & Anttila, M. 1995. Criticality safety calculations at VTT Energy. In: Proceedings of the 7th Nordic Reactor Physics Meeting "Reactor physics calculations in the Nordic countries", Espoo 8 - 9 May 1995. Espoo: Technical Research Centre of Finland. Pp. 124 - 133. (VTT Symposium 159.)

Roine, T., Anttila, M., Höglund, R. & Solala, M. 1995. Validation of CASMO/SIMULATE code package for TVO boiling water reactors. Proceedings of the ANS Topical Meeting: International Conference on Mathematics and Computations, Reactor Physics, and Environmental Analyses, Portland, OR April 30 - May 4 1995. La Grange Park, IL: American Nuclear Society. Vol. 2. Pp. 1058 - 1067.

Räty, H.1995. Application of TRAB to BWR transient and accident analyses. In: Proceedings of the 7th Nordic Reactor Physics Meeting "Reactor physics calculations in the Nordic countries", Espoo 8 - 9 May 1995. Espoo: Technical Research Centre of Finland. Pp. 239 - 249. (VTT Symposium 159.)

Siltanen, P., Antila, M. & Kyrki-Rajamäki, R. 1995. Studies of core response in local boron dilution accidents. In: Proceedings of the 7th Nordic Reactor Physics Meeting "Reactor physics calculations in the Nordic countries", Espoo 8 - 9 May 1995. Espoo: Technical Research Centre of Finland. Pp. 290 - 302. (VTT Symposium 159.)

Siltanen, P. & Kyrki-Rajamäki, R. 1995. AER working group D on VVER safety analysis, minutes of the meeting in Espoo, Finland 17 - 19 May 1995. Proceedings of the 5th Symposium of the AER (Atomic Energy Research for Investigating Neutron Physics and Thermohydraulics Problems of Reactor Safety). Dobogókö, Hungary 5 - 19 October 1995. Budapest: KFKI Atomic Energy Research Institute. Pp. 17 - 23.

Tanskanen, A.1997. Generating high-temperature neutron data for MCNP4A with NJOY94 from ENDF/B-VI.3 and JEF-2.2. 8th Nordic Reactor Physics Meeting "Reactor Physics Calculations in the Nordic Countries", Kjeller, Norway 13 - 14 March 1997.

Vanttola, T., Rajamäki, M., Kyrki-Rajamäki, R., Tiihonen, O., Hänninen, M. & Miettinen, J. 1996. Methodology, status and plans for development and assessment of Finnish reactor dynamics codes and APROS code. OECD/CSNI Workshop on Transient Thermal Hydraulic & Neutronic Codes Requirements, Annapolis, MD 5 - 8 November 1996. (invited paper)

Wasastjerna, F. 1995. Neutron fluence calculations for Loviisa reactor vessel PTS considerations. STUK-IAEA seminar "Transient and accident analyses and codes validation for WWER-440 reactors", Helsinki 13 - 17 November 1995.

Wasastjerna, F. 1995. Pre-processors in the REPVICS System: TOPICS and PVIS-3". OECD/NEA Seminar on 3D Deterministic Radiation Transport Computer Programs: Features, Applications and Perspectives, OECD Chateau de la Muette Paris XVI 2 - 3 December 1996.

Research institute reports

Höglund, R. (ed.). 1995. Proceedings of the 7th Nordic Reactor Physics Meeting "Reactor physics calculations in the Nordic countries", Espoo 8 - 9 May 1995. Espoo. VTT Symposium 159. 353 p. + app. 2 p.

Kyrki-Rajamäki, R. 1995. Three-dimensional reactor dynamics code for VVER type nuclear reactors. Espoo. VTT Publications 246. 51 p. + app. 80 p. (DTech thesis, Helsinki University of Technology.)

Others

Tanskanen, A. 1997. Generation of program-wise reactor physics data libraries from the evaluated nuclear data libraries. Espoo: 58 p. (MScTech thesis, Helsinki University of Technology, in Finnish.)

Kyrki-Rajamäki, R. (ed.). 1997. VTT Energy Nuclear Newsletter. In focus: BWR safety and fuel performance. Espoo: VTT Energy. 4 p.

Peltonen, J.1996. Present view on the Chernobyl reactor accident. ATS Ydintekniikka, vol. 25, nro 1, pp. 2 - 5. (in Finnish.)

Stenius, T. 1995. New UO_2 conductance and heat capacity models of HEXTRAN and TRAB. Technical report RFD-1/95. 5 p.

Stenius, T. 1995. New UO_2 conductance and heat capacity models of HEXTRAN. Technical report RFD-2/95. 7 p + 8 app.

Wasastjerna, W. 1995. Learning to Use MCNP4.2 for Fusion Neutronics Calculations. Technical report RFD-4/95. 33 p. (funded by FFUSION project).

Kyrki-Rajamäki, R. 1995. Time-dependent solution of the nodal neutronics equations in the HEXTRAN code. Technical report RFD-7/95. 10 p.

Roine, T., Anttila, M., Höglund, R. & Solala, M. 1995. Validation of CASMO/SIMULATE code package for TVO boiling water reactors. Technical report RFD-9/95. 10 p.

Wasastjerna, W. 1995. CROCO, version 2.0. A program to collapse three-dimensional group constants to one dimension. Technical report RFD-10/95. 56 p.

Piponius, T. 1995. Numerical simulation of cross flows between sub-channels in a rod bundle and changes caused by mixing in the conditions of a hot channel. Technical report RFD-20/95. 63 p. (in Finnish.)

Piponius, T. 1995. Changes made in the COBRA-IV code. Technical report RFD-1/96. 8 p. (in Finnish.)

Piponius, T.1996. Comparisons between COBRA and TRAB. Technical report RFD-2/96. 36 p. (in Finnish.)

Räty, H. 1996. Fuel heat conductivity and heat capacity dependencies in TRAB version 3.5 and HEXTRAN version 2.7. Technical report RFD-3/96. 5 p.

Vanttola, T. 1996. KOLMED plotting code for 3-dimensional presentation of surfaces and points. Revised version. Replaces report RFD-2/92. Technical report RFD-4/96. 13 p. (in Finnish.)

Räty, H. 1996. Film boiling models in TRAB version 3.5. Technical report RFD-5/96. 5 p.

Räty, H. 1996. Reactor physics and dynamics methods including thermal hydraulic (DYNAMIC). Annual Report for 1995. Technical report RFD-9/96. 37 p + app. 2 p.

Kaloinen, E. 1996. Calculation of moderator coefficients in HEXBU-3D/MOD5. Technical report RFD-11/96. 5 p.

Kaloinen, E. 1996. Input instructions of HEXBU-3D/MOD5. Technical report RFD-12/96. 25 p.

Kaloinen, E. 1996. CRFIT, a code for creation of input file of two-group diffusion parameters and fitting coefficients for HEXBU-3D/MOD5. Technical Reports RFD-13/96. 9 p.

Peltonen, J. 1996. Calculations for fuel loading patterns of Triga core for a BNCT unit with the MCNP model. Technical report RFD-14/96. 4p. + app. 4 p. (In Finnish.)

Anttila, M. 1996. The fuel assembly program CASMO-HEX. Input Manual. Technical report RFD-17/96. 46 p.

Peltonen, J. 1996. Test of CASMO-4 for RBMK reactors, preliminary 1-D CLU model. Technical Report RFD-19/96. 5 p.

Wasastjerna, F. 1996. Neutron flux calculations for the ICRH vacuum windows in ITER. Part 1. Technical Report RFD-20/96. 16 p. (funded by FFUSION project.)

Wasastjerna, F. 1996. Neutron flux calculations for the ICRH vacuum windows in ITER. Part 2. Technical Report RFD-21/96. 9 p. (funded by FFUSION project.)

Wasastjerna, F. 1996. MCNP4A Calculations of the neutron and gamma flux in and near an ITER equatorial port containing an ICRH array. Technical Report RFD-22/96. 21 p. (funded by FFUSION project.)

Wasastjerna, F. 1996. Effects of an auxiliary shield on the neutron flux in the equatorial ports of ITER. Technical Report RFD-23/96. 6 p. (funded by FFUSION project.)

Wasastjerna, F.1996. Calculating the flux in an annular gap in a black material. Technical Report RFD-24/96. 9 p. (funded by FFUSION project.)

Wasastjerna, F. 1996. The dose rate from an activation product source behind a concrete shield of 1.2 m thickness containing a duct of 23 cm diameter. Technical Report RFD-25/96. 10 p. (funded by FFUSION project.)

Wasastjerna, F. 1996. BENCHMARK: Dose rate estimate around heat exchanger in FW/SB primary coolant loop. calculations at ITER JCT Garching Using MCNP4A and FENDL-1. Technical Report RFD-26/96. 8 p. (funded by FFUSION project.)

Wasastjerna, F.1996. Calculation of the heating rate in the lower outboard quadrant of the toroidal field coils of ITER due to neutrons streaming through the divertor port. Technical Report RFD-27/96. 12 p. (funded by FFUSION project.)

Wasastjerna, F. 1996. Work done as visiting home team personnel at ITER Joint Work Site Garching, 1996-03-11...06-07. Technical Report RFD-28/96. 4 p. + app. 8 p. (funded by FFUSION project.)

Narumo, T. & Rajamäki, M. 1996. Wall and interfacial friction terms of a dynamic whole-range two-fluid SFAV-model applicable to BWR-conditions. Technical Report RFD-29/96. 20 p.

Peltonen, J. 1996. Effect of the fundamental mode calculation of CASMO-4 on the results in hexagonal and square lattice geometries. Technical report RFD-30/96. 4 p. + app. 7 p. (in Finnish.)

Kaloinen, E. 1996. Test calculations for the nodal model of TRAB-3D. Part 2. Technical Report RFD-31/96. 7 p.

Laine, H. 1996. Calculation of test results in development of a solution algorithm based on piecewise linear interpolation for hyperbolic partial differential equations. Technical Report RFD-32/96. (In Finnish.)

Wasastjerna, F. 1996. Pre-prosessors in the REPVICS system: TOPICS and PVIS-3. Technical Report RFD-33/96. 7 p.

Wasastjerna., F. 1996. VENUS-1 Two-dimensional benchmark on ex-core dosimetry computations: Finnish contribution. Technical Report RFD-34/96. 40 p.

Wasastjerna, F. 1996. Rough estimate of radiation levels for the ITER in-vessel viewing system. Technical Report RFD-35/96. 4 p. (funded by FFUSION project.)

Heino, J. 1996. Development of a computer program for optimization of control rod patterns in a boiling water reactor. Technical Report RFD-36/96. 23 p. + app. 23 p. (In Finnish.)

Tukiainen, V. & Kaloinen, E. 1996. Calculations of NEACRP 3-D LWR core transient benchmark with three-dimensional dynamics code TRAB-3D. Technical Report RFD-37/96. 13 p + app. 67 p.

Höglund, R. & Solala M. 1996. ICFM for a BWR using the CORFU core designer. Technical Report RFD-38/96. 8 p.

Kaloinen, E. 1997. Test calculations for the RBMK option of the nodal flux modelling in TRAB-3D. Technical Report RFD-2/97. 8 p.

Heino, J. 1997. On the accuracy of CORFU2 and changes made in the code. Technical Report RFD-3/97. 5 p + app. 8 p. (in Finnish.)

Narumo, T. 1997. Study on level of complexity required for two-fluid model to predict correctly two-phase flow wave propagation phenomena. Technical Report RFD-4/97. 18 p. To be presented at and published by NURETH-8, Eight International Topical Meeting on Nuclear Reactor Thermal Hydraulics "New horizons in Nuclear Reactor Thermal Hydraulics", Kyoto, September 30 - 4 October 1997.

Thermal Hydraulic Experiments and Analyses (TEKOJA) and Passive Safety Injection Experiments (PAHKO)

Conference Papers

Munther, R., Vihavainen, J. & Kouhia, J. 1995. Analyses of PACTEL passive safety injection tests with RELAP5 code. IAEA Technical Committee Meeting ”Progress in Design, Research & Development and Testing of Safety Systems for ACWRs”. Piacenza, Italy 16 - 19 May 1995.

Korteniemi, V., Haapalehto, T., Virtanen, E., Aaltonen, T., Tuunanen, J. & Kouhia, J. 1995. Analysis of horizontal steam generator loss of feedwater experiments. ASME Fluids Engineering Annual Conference & Exhibition. The Westin Resort, Hilton Head Island, South Carolina 13 - 18 August 1995.

Korteniemi, V., Haapalehto, T. & Puustinen, M.1995. Reflooding and boil-off experiments in a VVER-440 like rod bundle and analyses with the CATHARE code. Proceedings of the 7th International Meeting on Nuclear Reactor Thermal-Hydraulics, NURETH-7, Saratoga Springs, NY 10 - 15 September 1995.

Munther, R., Kalli, H. & Kouhia, J. 1995. Condensation during gravity driven ECC: experiments with PACTEL. Proceedings of the 7th International Meeting on Nuclear Reactor Thermal-Hydraulics, NURETH-7, Saratoga Springs, NY 10 - 15 September 1995.

Raussi, P., Kainulainen, S. & Kouhia, J. 1995. Natural circulation in a VVER geometry: experiments with PACTEL facility and CATHARE simulation. Proceedings of the 7th International Meeting on Nuclear Reactor Thermal-Hydraulics, NURETH-7, Saratoga Springs, NY 10 - 15 September 1995.

Kalli, H., Miettinen, A., Purhonen, H., D'Auria, F., Frogheri, M. & Leonardi, .1995. Quantitative code accuracy evaluation of ISP33. Proceedings of the 7th International Meeting on Nuclear Reactor Thermal-Hydraulics, NURETH-7, Saratoga Springs, NY 10 - 15 September 1995.

Virtanen, E., Haapalehto, T. & Kouhia, J. 1995. Analysis of steam generator loss-of-feedwater experiments with APROS and RELAP5/MOD3.1 computer codes. Proceedings of the 7th International Meeting on Nuclear Reactor Thermal-Hydraulics, NURETH-7, Saratoga Springs, NY 10 - 15 September 1995.

Purhonen, H., Kouhia, J. & Kalli, H.1995. ISP33 Standard problem on the PACTEL facility. Proceedings of the 7th International Meeting on Nuclear Reactor Thermal-Hydraulics, NURETH- 7, Saratoga Springs, NY 10 - 15 September 1995.

Tuunanen, J., Sonnenburg, H. & Palazov, V. 1995. UPTF small break experiments: evaluation of condensation processes in UPTF tests A6 and A7. NURETH7, Saratoga Springs, NY 10 - 15 September 1995.

Munther, R., Vihavainen, J. & Kouhia, J. 1995. Simulation of PACTEL gravity driven core cooling tests with APROS(2.11) and RELAP5/MOD3.1 codes. Proceedings of the

ANS International Topical Meeting of Safety of Operating Reactors, Seattle, WA 17 - 20 September 1995.

Riikonen, V., Puustinen, M. & Kouhia, J. 1995. RELAP5/MOD3.1 & APROS 3.0 analysis of SBLOCA in scaled VVER-440 geometry. Eleventh Proceedings of Nuclear Thermal Hydraulics, 1995 ANS Winter Meeting, San Francisco, CA 29 October - 2 November 1995.Pp. 141 - 148.

Tuunanen, J., Munther, R. & Vihavainen, J.1996. PACTEL experiments for investigation of passive safety injection systems of advanced light water reactors. The 4th International Conference on Nuclear Engineering (ICONE-4), New Orleans, LA 10 - 14 March 1996.

Tuunanen, J. 1996. Experimental program on PACTEL for the investigation of passive safety injection systems of ALWRs. Jahrestagung Kerntechnik ‘96. Mannheim, Rosengarten 21. - 23. May 1996.

Ohvo, J., Tuomainen, M., Kalli, H. & Tuunanen, J. 1997. UPTF Loop sseal experiments and their simulaton with CATHARE and RELAP. ICONE-5 - International Conference on Nuclear Engineering, Nice, France 25. - 29.May 1997

Others

Purhonen, H. 1995. International standardproblem ISP33 on PACTEL -facility modelling VVER-type nuclear power plant. Lappeenranta: (Licentiate thesis, Lappeenranta University of Technology, in Finnish.)

Ritonummi, T., Kalli, H., Purhonen, H. & Kouhia, J. 1995. Nuclear plant safety research is advancing. Fysiikka tänään -Journal, nro 1. (in Finnish.)

Purhonen, H., Lundström, P. & Kymäläinen, O., 1995. Experimental support for Loviisa VVER- 440 accident management development, Nuclear Europe Wordscan, nro 9-10, pp. 52 - 53.

Tuunanen, J. & Riikonen, V. 1995.Test specification of PACTEL natural circulation tests in low primary pressure. Lappeenranta: VTT Energia, TEKOJA 1/95.

Riikonen, V. 1995. Steam generator multiple tube rupture experiments on the PACTEL facility. Lappeenranta: VTT Energia, TEKOJA 5/95

Bánáti, J. 1995. Assessment of RELAP5/MOD3.1 against the 4th IAEA standard problem excercise performed on PMK-2 facility. Espoo: VTT Energia, TEKOJA 6/95.

Tuunanen, J. 1995. Proposal for a test series. primary system depressurization tests on PACTEL. Lappeenranta: VTT Energia. TEKOJA 7/95.

Kouhia, J. 1995. OECD support group on VVER code validatation matrix). Lappeenranta: VTT Energia, TEKOJA 8/95. (Matkakertomus.)

Tuunanen, J. & & Kouhia, J. 1995. Modification of the PACTEL test facility for the Gravity Driven Emergency Cooling Tests (GDE-11, -12, -13 and -14).Lappeenranta: VTT Energia. TEKOJA 11/95.

Tuunanen, J. 1995.Assessment of RELAP5/mod3.1 coe agaunst PACTEL passive safety injection test GDE-11.Lappeenranta: VTT Energia. TEKOJA 12/95.

Semken, S. & Tuunanen, J.1996. Experimental data report, PACTEL natural circulation experiments in low primary pressure. Lappeenranta: VTT Energia.TEKOJA 4/95.

Riikonen, V. 1996. RELAP5/mod3.2 Analyses of the steam generator mutible tube rupture experiments (PSL02 and PSL04) on the PACTEL facility. Lappeenranta: VTT Energia. TEKOJA 2/96.

Tuunanen, J. & Semken, S. 1996. Proposal for test series. The effects of pressure on hot leg loop seal behaviour. Lappeenranta: VTT Energia, TEKOJA 3/96.

Kouhia, J., Haapalehto, T. & Kagitcibasi, B. 1996. Effect of steam generator inclination on SBLOCA in VVER-440 geometry. Lappeenranta: VTT Energia. TEKOJA 4/96.

Riikonen, V. & Semken, S.1996. steam generator mutible tube rupture experiments (PSL05, PSL06 and PSL07) on the PACTEL facility. Quick Look Report. Lappeenranta: VTT Energia. TEKOJA 5/96.

Puustinen, M. 1996. Small break loss-of-coolant accident experiments SBL-30, SBL-31, SBL-32 and SBL-33 in the PACTEL facility. Quick Look Report. Lappeenranta: VTT Energia. TEKOJA 6/96.

Riikonen, V. 1996. Steam generator multiple tube rupture experiments on the PACTEL facility with the new steam generators. Lappeenranta: VTT Energia. TEKOJA 8/96.

Tuunanen, J. 1996. Test specification report. passive safety injection tests GDE31-GDE35. Lappeenranta: VTT Energia, PAHKO 1/96.

Tuunanen, J. 1996. Quick look report. passive safety injection experiments GDE-21, GDE-22, GDE-23, GDE-24 and GDE-25. Lappeenranta: VTT Energia. PAHKO 2/96.

Tuunanen, J., Lillington, J., D'Auria, F. & Kalli, H. 1996. Review of experimental work and code application on the simulation of passive safety injection systems. Lappeenranta: VTT Energia. PAHKO 3/96.

Tuunanen, J. 1996. Experimental data report. passive safety injection experiments GDE-21, GDE-22, GDE-23, GDE-24 and GDE-25.Lappeenranta: VTT Energia. PAHKO 4/96.

Tuunanen, J. 1996. Progress report of the European Commission Nuclear Fission Safety Program Project ”Assessment of passive safety injection systems of advanced light water reactors”.Lappeenranta: VTT Energia. PAHKO 5/96.

Semken, S. 1996. Quick look report. pressure drop measurements of the passive safety injection system lines of PACTEL. Lappeenranta: VTT Energia PAHKO 6/96.

Tuunanen, J. & Vihavainen, J. 1996. Test specification report. passive safety injection tests GDE31-GDE35. Lappeenranta: VTT Energia. PAHKO 7/96.

Semken, S.1996. Quick look report. pressure drop measurements of the passive safety injection system lines of PACTEL for GDE-31 through GDE-35. Lappeenranta: VTT Energia. PAHKO 8/96.

Vihavainen, J. 1997. Analysis of passive safety injection experiment GDE-24 with APROS 4.02 Code. Lappeenranta: VTT Energia. PAHKO 9/96.

Vihavainen, J.& Kalli, H. 1996. On the applicability of present computer codes for the calculation of the future NPPs. Lappeenranta: LTKK. (Research Report EN B-103.) ISBN 951-764-068-4.

Tuunanen, J. 1997.Quick look report. Primary system depressurization experiments CPR-11 and CPR-12. Lappeenranta: VTT Energia, TEKOJA 1/97.

Tuunanen, J. 1997. Annual progress report of the European Commission Nuclear Fission Safety Program Project "Assessment of passive safety injection systems of advanced light water reactors". Lappeenranta: VTT Energy. PAHKO 1/97.

Tuunanen, J. & Puustinen, M. 1997. Quick look report. Passive safety injection experiments GDE-31, GDE-32, GDE-33, GDE-34 and GDE-35. Lappeenranta: VTT Energy. PAHKO 2/97.

Tuomainen, M. & Tuunanen, J. 1997. UPTF loop seal tests and their RELAP simulation. Lappeenranta: LTKK. (Tutkimusraportti EN A-40.) ISBN 951-764-130-3.

Munther, R., Vihavainen, J., Kalli, H., Kouhia, J. & Riikonen, V. 1997. RELAP5 analysis of gravity driven core cooling experiments with PACTEL Nuclear Technology, September 1997 (to appear.)

Severe Accident Management (VAHTI) and
Reactor Accidents' phenomena and Simulation (ROIMA)

Publications in scientific journals

Lindholm, I.; Pekkarinen, E. & Sjövall, H. 1995. Evaluation of reflooding effects on an overheated boiling water reactor core in a small steam-line break accident using MAAP, MELCOR and SCDAP/RELAP5 codes. Nuclear Technology, vol. 112, nro 1, pp. 42 - 57.

Conference papers

Artnik, J., Caroli, C., Generino, G., Parozzi. F., Pini, A., Sairanen, Risto & Turland, B.D. 1995. Molten pool behaviour calculations for reactor cases. FISA-95 Symposium. EU Research on Severe Accidents, Luxembourg 20 - 22 November 1995.

Berg, Ø., Sørenssen, A., Lindholm, I., Miettinen, J., Pekkarinen, E., Sjövall, H. & Koski, S. 1995. Analysis of severe accident phenomenology and need for computerized accident management support. OECD/CSNI/SESAM Specialist meeting on Severe Accident Management Implementation, Niantic. Connecticut, 12 - 14 June 1995.

Jokiniemi, J., Mäkynen, J., Kauppinen, E., Tuomisto, H. & Hongisto, O. 1995. LWR containment aerosol experiments at Victoria facility. Proceedings of the 1995 European Aerosol Conference, Helsinki, Finland 18 - 22 September 1995.In: J. Aerosol Sci.vol. 26, suppl.1, pp. 713 - 714.

Karppinen, I. 1995. Modelling studies of horizontal steam generator PGV-1000 with CATHARE. Third International Seminar on Horizontal Steam Generators. Lappeenranta 18 - 20 October 1994. Lappeenranta: Lappeenranta University of Technology. Pp. 273 - 290. (Scientific Publications 43.)

Lindholm, I.; Pekkarinen, E.; Nilsson, L. & Sjövall, H. 1995. Analysis of core coolability under reflooding conditions for Nordic boiling water reactors. TOPSAFE '95. International EastWest TOPical Meeting. Safety of Operating Nuclear Power Plants. Budapest 24 - 27 September 1995. Berne: European Nuclear Society (ENS). Pp. 183 - 188. (Transactions, volume II. Poster Papers.) isbn 3-9520691-1-6

Miettinen, J. 1995. SMABRE: A fast running thermal-hydraulic code for transient and accident analyses. STUK-IAEA Seminar "Transient and accident analyses and codes validation for WWER-440 reactors". Helsinki 13 - 17 November 1995.

Mäkynen, J., Jokiniemi, J., Ahonen, P., Kauppinen, E. & Zilliacus, R. 1995. Experimental studies on hygroscopic CsOH and CsI and inert Ag aerosol behaviour in LWR containment conditions. Proceedings of the 1995 European Aerosol Conference. Helsinki, Finland 18 - 22 September 1995. In: J. Aerosol Sci.,vol. 26, suppl.1. Pp. 85 - 86.

Mäkynen, J., Jokiniemi, J. & Kauppinen, E. 1995. CsOH, CsI and Ag aerosol behavior in LWR containment conditions. In: Abstracts for the American Association for Aerosol Research, 14th Annual Meeting, Pittsburgh, PA 9 - 12 October 1995. P. 66.

Mäkynen, J., Jokiniemi, J., Ahonen, P., Kauppinen, E., Tuomisto, H. & Routamo, T. 1996. Experimental studies on LWR containment aerosol behaviour at VICTORIA facility. Proceedings of the 1996 European Aerosol Conference, Delft 9 - 12 September 1996. In: J. Aerosol Sci., vol. 27, suppl.1. pp. 469 - 470.

Mäkynen, J., Jokiniemi, J., Ahonen, P., Kauppinen, E., Tuomisto, H. & Routamo ,T.1996. Experimental studies on LWR aerosol behaviour at VICTORIA model containment. In: Abstracts of the American Association for Aerosol Research 15th Annual Meeting, Orlando, FL 14 - 18 October 1996. P. 337.

Norrman, S. 1995. Development and validation of APROS plant analyzer.”Transient and accident analyses and codes validation for WWER-440 reactors”. Helsinki 13 - 17 November 1995.

Puska, E.K. & Kontio, H. 1995. Three-dimensional core in APROS plant analyzer, Proceedings of the International Conference on Mathematics and Computations, Reactor Physics, and Environmental Analyses, Portland, OR 30 April - 4 May 1995. La Grange Part: IL: American Nuclear Society. Vol. I, pp. 264 - 273. (Funded by the APROS reactor models project). ISBN 0-89448-198-3

Puska, E.K., Miettinen, J., Hänninen, M., Kontio, H. & Honkoila, K. 1995. APROS simulation system for nuclear power plant analysis, Proceedings of ICONE-3, The 3rd JSME/ASME Joint International Conference on Nuclear Engineering, Kyoto, Japan 23 - 27 April 1995. Tokyo: The Japan Society of Mechanical Engineers. Vol. 1, pp. 83 - 88. (Funded by the APROS reactor models project).

Puska, E.K., Hämäläinen, A. & Kontio, H. 1995. Visualization in large plant analyzer applications, Proceedings of the 1995 Simulation Multiconference, High Performance Computing Symposium, Phoenix, AZ 9 - 13 April 1995. San Diego, CA: The Society for Computer Simulation.. Pp. 429 - 434. ISBN 1-56555-078-1(Funded by the APROS reactor models project).

Puska, E.K. & Kontio, H. 1995. APROS simulation environment in nuclear plant simulation, 1995 Summer Computer Simulation Conference, Ottawa, Ontario 24 - 26 July 1995. San Diego, CA: The Society for Computer Simulation, ISBN 1-56555-081-1, pp. 703 - 708. (Funded by the APROS reactor models project**).**

Puska, E. K., Norrman, S., Miettinen, J. & Hänninen, M. 1997. APROS BWR model for transient analysis. 5th International Conference on Nuclear Engineering (ICONE-5), Nice, France. 26 - 30 May 1997.

Sairanen, R. 1995.Quality assurance of accident analyses. ”Transient and accident analyses and codes validation for WWER-440 reactors”. Helsinki 13 - 17 November 1995.

Vanttola, T., Rajamäki, M., Kyrki-Rajamäki, R., Tiihonen, O., Hänninen, M. & Miettinen, J. 1996. Methodology, status and plans for development and assessment of Finnish reactor dynamics codes and APROS code. OECD Workshop on Transient Thermal Hydraulic Code Requirements, Annapolis, MD 5 - 8 November 1996.

Ylijoki, J., Hänninen, M. & Puska, E.K.1995. APROS in conventional and nuclear power plant simulation, Proceedings of SIMS'95, Lyngby, Denmark 28 - 30 June 1995.Pp. 54 - 64. (Funded by the APROS reactor models project.)

Yrjölä, V. 1995. Results of questionnaire for the needs of measured data for the steady-state calculations. Third International Seminar on Horizontal Steam Generators, Lappeenranta. 18 - 20 October 1994. Lappeenranta: LTKK. Pp. 56 - 63. (Tieteellisiä julkaisuja 43.)

Research institute reports

Ikonen, K.1997. Reaktoripaineastian pohjan ja läpivientien kuumeneminen sydämen sulamisonnettomuuksissa. Espoo: VTT Energia. 120 s. + liitt. 18 s. (VTT Tiedotteita - Research Notes 1831.) ISBN 951-38-5108-7.

Jokiniemi, J.; Kilpi, K.; Lindholm, I.; Mäkynen, J.; Pekkarinen, E.; Sairanen, R. & Silde, A. 1995. Severe accident phenomena. Espoo VTT Energy. 103 p. (VTT Tiedotteita 1628). (in Finnish.)

Mäkynen, J. & Jokiniemi J. (eds) 1995. CSNI/PWG/FPC AHMED code comparison exercise. Le Seine, Saint Germain. OECD/NEA/CSNI/R(95)23. 80 p.

Højerup, F. Miettinen, J. Nilsson, L., Puska, E.K., Sjövall, H., Anttila, M. & Lindholm, I. 1997. On recriticality during reflooding of a degraded boiling water reactor core. February 1997. (NKS/RAK-2(97)TR-A3.) ISBN 87-7893-010-3.

Others

Holmström, H. & Sairanen, R. 1995. Severe accident research. ATS Ydintekniikka, vol. 24, nro 2, pp. 3 - 5. (in Finnish.)

Lindholm, I., Nilsson, L., Pekkarinen, E. & Sjövall, H. 1995 Coolability of degraded core under refooding conditions in Nordic boiling water reactors. Final Report NKS/RAK-2(95)TR-A1, September 1995.

Mäkynen, J., Jokiniemi, J., Ahonen P. & Kauppinen, E. 1995 AHMED tests on hygroscopic and inert aerosol behaviour in LWR containment conditions. Commission of the European Communities. Reinforced Concerted Action on Reactor Safety. Report ST(95)-P174.

Miettinen, J. 1995. Main features of the SMABRE code, Version 4.2. Technical Report VAHTI-1/95. 20 p.

Miettinen, J. & Hämäläinen, A. 1995. SMABRE-4.2 validation: ROSA-IV experiment SB-CL-18, 5 % SBLOCA. Technical Report VAHTI-2/95. 14 p.

Miettinen, J. & Hämäläinen, A. 1995. SMABRE-4.2 analysis: SPES experiment SP-FW-02, Loss of feedwater, ISA-22. Technical Report VAHTI-3/95. 12 p.

Miettinen, J. & Hämäläinen, A. 1995. SMABRE-4.2 validation: The Doel plant steam generator tube rupture, ISP-20. Technical Report VAHTI-4/95. 12 p.

Miettinen, J. & Hämäläinen, A. 1995.SMABRE-4.2 validation: LOBI-MOD2 experiment A2-81, 1 % SBLOCA, ISP-18. Technical Report VAHTI-5/95. 10 p.

Miettinen, J. & Hämäläinen, A. 1995. SMABRE-4.2 validation: The BWR plant SBLOCA in piper one facility, ISP-21. Technical Report VAHTI-6/95. 11 p.

Miettinen, J. & Hämäläinen, A. 1995. SMABRE-4.2 validation: LOFT 2.5 % SBLOCA experiment L3-6, ISP-11. Technical Report VAHTI-7/95. 15 p.

Taalikka, M. 1995. APROS 3.0 calculation: the Marviken full scale containment experiment BP 18. Technical Report VAHTI-13/95. 25 p.

Yrjölä, V. 1995. Short guidelines for the installation of the NPA system 1.34 on the HP 9000/735 workstation. Technical Report VAHTI-14/95. 10 p. + app. 3 p.

Yrjölä, V. 1995. Short guidelines for the installation of the RELAP5/MOD3.1 on the HP 9000/735 workstation. Technical Report VAHTI-15/95. 12 p. + app. 2p.

Silde, A. 1995. A summary of the hydrogen transport and deflagration experiments at the HDR facility. Technical Report VAHTI-18/95. 19 p. + app.

Lindholm, I.1995. NKS/RAK-2 annual report 1995 and project plan for 1996-97. Technical Report VAHTI-24/95. 20 p.

Karppinen, I. 1995. Modelling of the steam generator collector break in VVER-1000 with CATHARE. Technical Report VAHTI-25/95. 12 p.

Karppinen, I. 1996. Modelling the Olkiluoto steam separators with RELAP5. Technical Report VAHTI-3/96. 12 p.

Ikonen, K. 1996. Calculating creep in thermal-elastic-plastic deformation and stress analysis. Technical Report VAHTI-6/96. 48 p.

Ikonen, K. 1996. Creep in thermo-elastic-plastic deformation and stress analysis. Technical Report VAHTI-8/96. 48s.

Lindholm, I. 1996. On core melt behaviour in the reactor pressure vessel lower head in TVO boiling water reactor. Technical Report VAHTI-9/96.

Karppinen, I. 1996. Interfacial friction test with APROS 4.0 6-equation model. Technical Report VAHTI-10/96.

Ikonen, K. & Lindholm, I. 1996. A numerical analysis for determination of the effective thermal conductivity of granular core debris. Technical Report VAHTI-11/96. 16 p.

Karppinen, I. 1996. Simulation of IVO full-scale loop seal experiment with APROS. Technical Report VAHTI-12/96.

Karppinen, I. 1996. Simulation of LOTUS air/water vertical annular flow experiment with APROS. Technical Report VAHTI-13/96.

Sairanen, R. 1997. High temperature oxidation of zirconium and stainless steel: description of oxidation models proposed for the APROS code system. ROIMA-2/97. 16 p.

Lindholm, I. 1997. NKS/RAK-2 annual report 1996 and project plan 1997. ROIMA-3/97. 24 p.

Ikonen, K, Kelppe, S. & Ranta-Puska, K. 1997. Mechanical analysis of a nuclear fuel element during transient thermal load. ROIMA-6/97 (PATRA-3/97).

Reliability and risk analyses (LURI)

Publications in scientific journals and books

Arjas, E. & Holmberg, J.1995. Marked point process framework for living probabilistic safety assessment and risk follow-up, Reliability Engineering and System Safety, vol. 49, pp. 59 - 73.

Wahlström, B. 1995. Modeling of man-machine systems; a challenge for systems analysis. In: Giampiero E.G. Beroggi, William A. Wallace (Eds.). Computer Supported Risk Management. Pp.61 - 76. Kluwer Academic Publishers, Dordrecht.

Holmberg, J.1996.Risk follow-up by probabilistic safety assessment - experience from the Finnish pilot study, Reliability Engineering and System Safety. .Elsevier Science Limited. Vol. 53, pp. 3 - 15.

Conference papers

Holmberg, J. & Arjas, E. 1995. Risk follow-up by probabilistic safety assessment using marked point process framework. In: Proc. of the 8th ESReDA Seminar on Reliability Data Analysis & Use, Helsinki 17 - 18 May 1995. Pp. 279 - 293.

Lehtinen, E. & Wahlström, B. 1995. Some ongoing Finnish research activities connected to NPP safety and operational performance. IAEA TC on Review of Safety Performance of Operational Nuclear Power Plants, Vienna 23 - 25 October 1995. (4-TC-955.)

Lehtinen, E., Wahlström, B. & Piirto, A. 1995. Management of safety through performance indicators for operational maintenance. International conference on advances in operational safety of nuclear power plants, Vienna 4 - 8. September 1995. (IAEA-CN-61/25.)

Pulkkinen, U.1995. A Bayesian framework for modelling expert judgments. In: Proc. of the 8th ESReDA Seminar on Reliability Data Analysis & Use, Helsinki 17 - 18 May 1995.

Pyy, P. & Reiman, L. 1995. Human and organisational behaviour and performance indicators - Finnish contribution to the IAEA programme on collection and classification of human reliability data for use in probabilistic risk assessment. IAEA coordinated research programme first meeting, Vienna 3 - 7 April 1995.

Wahlström, B. 1995. Some cultural flavors of safety culture. ANS - IAEA - OECD/NEA International Topical Meeting on Safety Culture in Nuclear Installations, Vienna 24 - 28 April 1995.

Andersson, K., Pyy, P., Holmgren, P. & Jacobsson, P.1996. Analysis of four sequences with different methodological approaches. NKS/RAK Seminar, Stockholm 23 - 24 January 1996.

Holmberg, J. 1996. A decision model for the optimization of the plant shutdown criteria and test intervals. In: Probabilistic Safety Assessment PSA ‘96, Park City, Utah, September 21 - October 3 1996. Pp. 355 - 362. La Grange Park, IL: American Nuclear Society.

Holmberg, J., Hukki, K., Norros, L., Pulkkinen, U. & Pyy, P. 1996. Contextual approaches to systems safety — analysis of decision making in an accident situation. In: P.C. Cacciabue & I.A. Papazoglou (eds.). Probabilistic Safety Assessment and Management '96— PSAM-III, Crete, Greece, 24 – 28 June 1996. Pp. 1030 – 1035. London: Springer Verlag.

Lehtinen, E., Piirto, A., Unga, E. & Wahlström, B. 1996. Performance indicators - a tool for monitoring NPP performance. IAEA specialist meeting on NPP performance and statistical analysis, Vienna 30 September - 2 October 1996.

Pulkkinen, U. & Pyy, P. 1996. A bayesian approach for modeling expert judgements. In: P.C. Cacciabue & I.A. Papazoglou (eds.). Probabilistic Safety Assessment and Management '96— PSAM-III, Crete, 24 – 28 June 1996. London: Springer Verlag

Pyy, P. & Wahlström, B.1996. Support systems for nuclear power plant shutdown and low power operation. The Enlarged Halden Programme Group meeting, Loen, Norway. 19 - 23 May 1996.

Wahlström, B. 1996. Safety work in Finland and in Sweden. NKS/RAK Seminar, Stockholm, 23 - 24 January 1996. (in Swedish.)

Wahlström, B. 1996. Systems safety in the high-tech industrial environments; technology and human reliability. Key note presentation at the International Symposium Work in the Information Society, Helsinki, Finland 20 - 22 May 1996.

Wahlström, B.1996. In control of nuclear safety. The Enlarged Halden Programme Group meeting, Loen, Norway 19 - 23 May 1996.

Wahlstrom, B. 1996. Safety culture and communication; A comparison between Finnish and Swedish practices in nuclear safety. SREDA Autumn Seminar / SRD Association Annual Conference - "Communicating Safety",Culham (Oxford) 22 - 23 October 1996.

Research institute reports

Holmberg, J. 1995. A decision model for the risk management of hazardous processes. Laxenburg: International Institute for Applied Systems Analysis. (Report WP-95-95.)

Lehtinen, E. 1995. A concept of safety indicator system for nuclear power plants. 26 p. + app. 106 p. Espoo: VTT. (VTT Research Notes 1646.)

Holmberg, J. & Pulkkinen, U. 1996. Use of decision analytic methods in nuclear safety — an international survey. Helsinki: Finnish Centre for Radiation and Nuclear Safety. (STUK-YTO-TR 116.)

Pulkkinen, U. & Huovinen, T. 1996. Model uncertainty in safety assessments. Helsinki: Finnish Centre for Radiation and Nuclear Safety. 34 p. (STUK-YTO-TR 95.)

Wahlström, B., Nyman, R &. Reiman, L. 1996. Comparison of the work of safety authority in Finland and in Sweden. (NKS/RAK-1(96)R7.) (in Swedish.)

Holmberg, J. A decision model for the risk management of hazardous processes. Espoo: Helsinki University of Technology, Systems Analysis Laboratory. 23 p. (Report A 65.)

Holmberg, J. A utility theoretic view on probabilistic safety criteria. 19 p. Espoo: Helsinki University of Technology, Systems Analysis Laboratory. (Report A 64.)

Pulkkinen, U. & Holmberg, J. A method for using expert judgment in PSA. 32 p. (STUK-YTO-129.)

Holmberg, J. 1997. Probabilistic safety assessment and optimal control of hazardous technological systems. A marked point process approach. Espoo. VTT Publications 305. 32 p. + app. 77 p. (DTech thesis, Helsinki University of Technology.)

Others

Lehtinen, E. Incorporation of the influences of organizational factors into PSA - a survey of the approaches and methodologies. 32 p. Espoo:VTT Automation, Industrial Automation. (TAU-5005.)

Pyy, P., Pulkkinen, U., Norros, L., Hukki, K. & Holmberg, J. Decision analytical method for sequence analysis - VTT Automation contribution to NKS/RAK-1 subproject 3 "Integrated sequence analysis". 9 p. Espoo: VTT Automation, Industrial Automation. (TAU-5010.)

Huovinen, T. & Pulkkinen, U. Effect of safety classification on the reliability of equipment - an application on motor operated valves. Espoo: VTT Automation (TAU-6002.) (in Finnish.)

Pulkkinen, U. 1996. A methodology for using expert judgment. Espoo: VTT Automation (Report TAU-6020/96.)

Pulkkinen, U. 1996. Use of expert judgment in Finland. Espoo: VTT Automation (Report TAU-6021.)

Pyy, P. & Pulkkinen, U. 1996. Need to develop aging models and analyses as a part of the LURI project. 9 p. Espoo: VTT Automation (TAU-6022/96.) (In Finnish.)

Pulkkinen, U. & Holmberg, J. 1997. A method for using expert judgment in PSA. 36 p. , (Report TAU-7001/97.) Espoo: VTT Automation.

Human Factors in NPP Operation (ORINT) publications 1995-1997

Publications in Scientific Journals and Books

Norros, L. An orientation-based approach to expertise. In: J.M. Hoc, P.C. Cacciabue & E. Hollnagel (Eds.) Expertise and technology: Cognition and human-computer cooperation. Hilsdale, NJ: Lawrence Erlbaum Associates 137-160)

Norros, L. 1996. System disturbances as spingboard for development of operators´expertise. In: Engström, Y. And Middleton, D. (eds.) Cognition and Communication at Work. Cambridge: Cambridge University Press. Pp. 159 - 176.

Conference Papers

Norros, L. 1995. Evaluation and development of NPP crews' decision-making. APA 103rd Annual Convention, New York 11 - 15 August 1995. (abstract.)

Norros, L. & Hukki, K. Contextual analysis of the operators' on-line interpretations of process dynamics. In: Proceedings of the 5th European conference on cognitive science apparoches to process control. Pp. 182 - 195. Espoo: Technical Research of Finland. (VTT Symposium 158.)

Holmberg, J., Hukki, K., Norros, L., Pulkkinen, U. & Pyy, P. 1996. Contextual approaches to systems safety — analysis of decision making in an accident situation. In: P.C. Cacciabue & I.A. Papazoglou (eds.) Probabilistic Safety Assessment and Management '96— PSAM-III, Crete, Greece 24 – 28 June 1996. London: Springer Verlag. Pp. 1030 – 1035.

Norros, L. & Hukki, K. 1996. Development of a simulator training method for Nuclear Power Plants. A paper presented at the 2nd International Conference on Human Factors-Research In Nuclear Power Operations ICNPO 2), Berlin 28 - 30 November 1996.

Norros, L. 1996 The VTT Automation approach to validation of process information systems. Report of the SKI seminar on ”Verifiering och Validering av Kontrollrumsändringar”, Stockholm 18 - 19 April 1996.

Research Institute Reports

Norros,L. (Ed.) Proceedings of the 5th European conference on cognitive science apparoches to process control. Espoo: Technical Research Centre of Finland. (VTT Symposium No 158.)

ANNEX C

Steering group and project reference groups of the RETU research programme in 1997

Steering group of the RETU research programme

DTech Lasse Reiman, Chairman	Finnish Centre for Radiation and Nuclear Safety (STUK)
MScTech Markku Friberg	Teollisuuden Voima Oy
DTech Timo Haapalehto	Ministry of Trade and Industry
Prof. Prof. Heikki Kalli	Lappeenranta University of Technology
Prof. Mikko Kara	VTT Energy
MScTech Matti Komsi	IVO Power Engineering Oy
Prof. Lasse Mattila	VTT Energy
DTech Rauno Rintamaa	VTT Manufacturing Technology

Reference group for the Transient Models of Nuclear Fuel (PATRA) and Calculation Methods of Reactor Physics and Dynamics (DYNAMIC) projects

MSc Keijo Valtonen, Chairman	Finnish Centre for Radiation and Nuclear Safety (STUK)
MScTech Matti Ojanen	STUK
MScTech Pertti Siltanen	IVO Power Engineering Oy
MScTech Martti Antila	IVO Power Engineering Oy
MScTech Risto Teräsvirta	IVO Power Engineering Oy
MScTech Seppo Koski	Teollisuuden Voima Oy
MScTech Esa Mannola	Teollisuuden Voima Oy
LicTech Ralf Lunabba	Teollisuuden Voima Oy
MScTech Pertti Aaltonen	VTT Manufacturing Technology
LicTech Risto Sairanen	VTT Energy

Reference group for the Thermal Hydraulic Experiments and Analyses (TEKOJA) and the Passive Safety Injection Experiments (PAHKO) projects

DTech Harri Tuomisto, Chairman	IVO Power Engineering Oy
DTech Juhani Hyvärinen	Finnish Centre for Radiation and Nuclear Safety (STUK)
MScTech Samuli Savolainen	Imatran Voima Oy
MScTech Markku Friberg	Teollisuuden Voima Oy
DTech Markku Rajamäki	VTT Energy
LicTech Olli Tiihonen	VTT Energy
MScTech Kari Santaoja	VTT Manufacturing Technology
MScTech Virpi Korteniemi	Lappeenranta University of Technology

Reference group for the Reactor Accident Phenomena and Modelling (ROIMA) project

MScTech Kalevi Haule, Chairman	Finnish Centre for Radiation and Nuclear Safety (STUK)
DTech Juhani Hyvärinen	STUK
DTech Harri Tuomisto	IVO Power Engineering Oy
MScTech Olli Kymäläinen	Imatran Voima Oy
MScTech Heikki Sjövall	Teollisuuden Voima Oy
MScTech Seppo Koski	Teollisuuden Voima Oy
DTech Rauno Rintamaa	VTT Manufacturing Technology
LicTech Pekka Pyy	VTT Automation
DTech Riitta Kyrki-Rajamäki	VTT Energy

Reference group for the Reliability and Risk Analyses (LURI) and Human Factors in NPP Operations (ORINT) projects

MScTech Reino Virolainen, Chairman	Finnish Centre for Radiation and Nuclear Safety (STUK)
MScTech Vesa Ruuska	STUK
DTech Jussi Vaurio	Imatran Voima Oy
MScTech Bob Mohsen	Imatran Voima Oy
MScTech Risto Himanen	Teollisuuden Voima Oy
Mr Markku Malinen	Teollisuuden Voima Oy
LicTech Risto Sairanen	VTT Energy
Prof. Veikko Rouhiainen	VTT Manufacturing Technology
Prof. Björn Wahlström	VTT Automation

Personnel of the Transient Models of Nuclear Fuel (PATRA)

Person	Tasks
MScTech Seppo Kelppe	RIA-analyses, SCANAIR-code development
MScTech Kari Ranta-Puska	Mechanics studies, ENIGMA-development
Mr Jan-Olof Stengård	FRAP-T -verification, program maintenance
LicTech Kari Ikonen	Mechanics studies

Personnel of the Calculation methods of reactor physics and dynamics (DYNAMIC) project

Person	Task
MScTech Hanna Räty	Project manager, development and testing of applying PLIM in the reactor dynamics codes (improved thermal hydraulics modelling); testing of TRAB-3D
MScTech Markku Anttila	Reactor Physics; OECD/NEA connections; NEANSC (science committee)
LicTech Randolph Höglund	Reactor Physics, Nordic connections
MScTech Elja Kaloinen	Development and testing of TRAB-3D
DTech Riitta Kyrki-Rajamäki	Development and testing of HEXTRAN-PLIM (including circuit model) and TRAB-3D; International cooperation on VVER safety, eg. VVER dynamics benchmarks
MScTech Timo Narumo	Development and validation of SFAV (new 6-equation thermal hydraulics model)
MScTech Jyrki Peltonen	Validation of hexagonal CASMO-4, application of MCNP (Monte Carlo -calculation), development of a new cross section model
DTech Markku Rajamäki	Development, testing and application of CFDPLIM and SFAV (thermal hydraulic models and accurate solution methods)

MScTech Aapo Tanskanen	Processing of nuclear data libraries (JEF2.2/NJOY); validation of criticality safety methods
DTech Timo Vanttola	Hot channel modelling
LicTech Frej Wasastjerna	Reactor Physics; RBMK, MCNP (Monte Carlo-calculation)

Personnel of the Thermal Hydraulic Experiments and Analyses (TEKOJA) and the Passive Safety Injection Experiments (PAHKO) projects

Person/VTT Energy	**Tasks**
DTech Jari Tuunanen	Project manager; experiment data analyses
LicTech Heikki Purhonen	Experiment data analyses
MScTech Jyrki Kouhia	Data aqusition system; instrumentation
MScTech Markku Puustinen	APROS analyses
MScTech Vesa Riikonen	Experimental work; experiment data analyses; computer system manager

Person/LTKK	
MSc Scott Semken	Experiment data analyses
MScTech Juhani Vihavainen	Experiment data analyses; APROS analyses
Mr Harri Partanen	PACTEL operation and maintenance
Mr Ilkka Saure	PACTEL operation and maintenance; instrumentation and control systems
Mr Hannu Pylkkö	PACTEL operation and maintenance

Personnel of the Reactor Accidents' Phenomena and Modelling (ROIMA) project

Person	**Tasks**
LicTech Risto Sairanen	Project manager. Development of severe accident models for the APROS code
MScTech Heikki Holmström	OECD/CSNI cooperation in thermal hydraulics
LicTech Kari Ikonen	Development of the PASULA code
PhD Jorma Jokiniemi	Fission product behaviour
MScTech Ismo Karppinen	Thermal hydraulic validation of the APROS code
MScTech Klaus Kilpi	Severe accidents, EC Melt-Vessel Interaction project
MScTech Ilona Lindholm	Severe accidents, core melt behaviour in RPV lower plenum
MScTech Jaakko Miettinen	Development of thermal hydraulic models
LicTech Jouni Mäkynen	Fission product behaviour, planning of experiments
MScTech Sixten Norrman	Thermal hydraulic validation of the APROS code
MScTech Esko Pekkarinen	Severe accidents, long term accident management
LicTech Eija-Karita Puska	Severe accidents, BWR recriticality
MScTech Ari Silde	Development of severe accident models for the APROS code
MScTech Vesa Yrjölä	Thermal hydraulics, RELAP5 analyses
MSTech Tuomas Valmari	Fission product behaviour studies
LicTech Kari Lehtinen	Fission product behaviour studies
MScTech Jouni Pyykönen	Fission product behaviour studies
MSTech Ari Auvinen	Fission product behaviour studies

Personnel of the Reliability and Risk Analyses (LURI) project

Person	Task
LicTech Pekka Pyy	Project manager, human reliability assessment, integrated sequence analysis
DTech Urho Pulkkinen	Assisstant project manager, expert judgment, integrated sequence analysis, probability theory and mathematics
LicTech Jan Holmberg	Decision analysis, integrated sequence analysis applications of reliability theory
DTech Kari Laakso	Analysis of maintenance feedback data
Prof. Björn Wahlström	Analysis of safety work and culture, risk impact of NPP modifications and backfittings
LicTech Miki Sirola	Accident management and PSA, Risk impact of NPP modifications and backfittings

Personnel of the Human Factors in NPP Operations (ORINT) project

Person	Tasks
PhD Leena Norros	Project manager, analysis of AWA-method measurement qualities, operators' mobilization of subjective resources, integrated safety analysis
MA Kristiina Hukki	Development of the taxonomy of process situations (PhD work), development of information presentation

ANNEX D

Academic degrees in 1995 - May 1997

List of degrees awarded and thesis publications

Doctor of Technology degrees:

Kyrki-Rajamäki, R. 1995. Three-dimensional reactor dynamics code for VVER type nuclear reactors. Espoo. VTT Publications 246. 51 p. + app. 80 p. (Helsinki University of Technology.)

Holmberg, J. 1997. Probabilistic safety assessment and optimal control of hazardous technological systems. A marked point process approach. Espoo. VTT Publications 305. 32 p. + app. 77 p. (Helsinki University of Technology.)

Licentiate of Technology degrees

Purhonen, H. 1995. International standardproblem ISP33 on PACTEL -facility modelling VVER-type nuclear power plant. Lappeenranta. (Lappeenranta University of Technology, in Finnish.)

Master of Science in Technology degrees:

Tanskanen, A. 1997. Generation of program-wise reactor physics data libraries from the evaluated nuclear data libraries. Espoo. 58 p. (Helsinki University of Technology, in Finnish.)

Strandell, C. 1997. Concepts of nuclear power safety and their relationships. An analysis model of the components and contensts of safety work. Espoo. (Helsinki University of Technology, in Swedish.)

ANNEX E

Project research plans for the years 1997 and 1998

- **Transient Models of Nuclear Fuel (PATRA)**
- **Calculation Methods of Reactor Physics and Dynamics (DYNAMIC)**
- **Thermal Hydraulic Experiments and Analyses (TEKOJA)**
- **Passive Safety Injection Experiments (PAHKO)**
- **Reactor Accidents' Phenomena and Simulation (ROIMA)**
- **Reliability and Risk Analyses (LURI)**
- **Reliability and Risk Analyses (LURI)**
- **Human Factors in NPP Operations (ORINT)**
- **Administration and Information Activities of the Research Programme (REHTI)**

Transient Models of Nuclear Fuel (PATRA)

Duration	1994 - 1997
Project manager	Mr. Kari Ranta-Puska
Institute	VTT Energy
Volume and funding -97	1.4 person y. 1.04 MFIM
Funding sources in 97 (MFIM)	KTM 0.400, STUK 0.150, Utilities 0.300, Other 0.190

Objectives

This far most of the transient and accident tests have been performed for fresh fuel. The performance of a high burnup fuel can, however, significantly differ from the performance of a fresh fuel. The uncertainties involved in the transient behaviour have recently raised serious concern, and many nuclear countries have launched new experimental and modeling projects on this subject. The objectives of the PATRA project is to acquire the essential knowledge for assuring the safety of a high burnup fuel in transient conditions and to update the corresponding fuel performance models.

Main results so far

A co-operation agreement in the field of nuclear safety and protection has been agreed between the French IPSN and VTT. Specific topics include fuel behaviour studies under reactivity initiated accidents (RIA) and SCANAIR computer code development. Pellet deformation from a barrel shape to diabolo shape during a CABRI-reactor RIA test was demonstrated with the multi-dimensional code EPFMD of VTT Energy. Later a 3-D creep formulation has been constructed and linked with the elastic-plastic calculation in EPFMD. A test case of long-term cladding creep showed encouraging agreement with the true behaviour. English edition of the summary documentation was completed and delivered.

SCANAIR has been acquired from IPSN and tested by using simple cases. A member of the PATRA project has started the work period (roughly one year) in Cadarache 16.9.1996.

In the international IAEA Coordinated Research Programme FUMEX, ENIGMA was proven one of the best performers among the current state-of-the-art fuel behaviour codes. The models for high burnup effects need to be improved, though. For example, the UO_2 thermal conductivity model obviously yields too low values at very high burnups.

Planned results 1997:

Part of the development of the multi-d code EPFMD aims at fuel rod applications during fast transients. The significance of fuel/clad creep in RIA tests made in CABRI reactor will be evaluated. The difference in calculated fuel performance under RIA when using a 1-d code and a 3-d formulation is checked.

The USNRC is making major modifications to their FRAP-T6 code. VTT has preliminarily agreed to take part in the testing and validation of the code during its further development, in exchange of an access to the new high-burnup version of the code once completed.

New submodels specific for high burnups will be implemented in ENIGMA in co-operation with Nuclear Electric. The models will be validated against Halden data and power reactor data (VVER, BWR). Another version of the code will be tuned for probabilistic analyses.

Detailed information on the performance of highly exposed fuel will be collected through international co-operation.

Planned results 1998:

New fuel types and changes in reactor operation continue to place new requirements on the fuel research and code development for both steady state and transients. The influence of fission gas swelling and clad material properties on the failure mechanism in RIA will be clarified by results from international test programmes and related analytical work. The versatility of the ENIGMA code will be improved.

Applications:

Reactivity transients that would be possible in Finnish reactors will be analysed and the knowledge on fuel performance under such transients will be improved. Use of the SCANAIR and, later, FRAP-T codes will be made.

The ENIGMA code with its updated high burnup models is used to evaluate the effects of the planned reactor modernisations. Probabilistic methodology will be applied to generate the frequency distributions of the essential fuel rod parameters.

Calculation Methods of Reactor Physics and Dynamics (DYNAMIC)

Duration	1995 - 1998
Project manager	Ms. Hanna Räty
Institute	VTT Energy
Volume and funding 97 - 98	97: 5.2 person y. 3.18 MFIM, 98: 5.2 person y. 3.2 MFIM
Funding sources in 97 (MFIM)	KTM 0.60, STUK 0.60, VTT 1,98

Objectives

The main objective of the DYNAMIC project is to upgrade and extend the reactor physics and dynamics code system for tasks that are related to increased requirements and recent safety concerns. Applicability of the reactor physics codes needs to be ensured also for high burnup fuel. New methods have to be developed for the production of reliable and validated input data for reactor dynamics codes and other nuclear applications. The Monte Carlo technique is used for complex systems. Separate reactor physics codes are integrated into a validated calculation system. In reactor dynamics the objectives are to complement and validate the calculation system for complex reactivity accidents, such as ATWS, boron dilution and BWR core stability. The thermal hydraulic models of the dynamics codes are improved by taking into use the numerically accurate solution method PLIM and the physically based two-phase flow model SFAV.

Main results so far

Calculation and application methods of **HEXTRAN** documented (DTech thesis). **CFDPLIM** improved and tested for dynamics applications. **TRAB-PLIM** tested with NEANSC dynamic BWR benchmarks for coupled core channel neutronics-two-phase hydraulics calculation and hot channel modelling. Test bench model for a **VVER circuit** utilizing PLIM completed. Nodal solution of the **three-dimensional square lattice dynamics code** TRAB-3D completed, good accuracy in test calculations. Dynamics testing against NEANSC three-dimensional dynamics benchmarks started, PWR CRE cases completed successfully. Improved **cross section** description planned. **SFAV-model** validated against measurements; complicated behaviour successfully modelled with SFAV-equations solved by CFDPLIM; scientific publications. Sub-channel code COBRA taken into use in **hot channel** modelling, results compatible with TRAB. Latest version of the nuclear data processing code **NJOY** installed and applied. **CASMO-HEX** input manual completed. **CASMO-4** installed, hexagonal validation started with Studsvik. A method for optimization of control rod patterns successfully implemented in **CORFU**, test loading designed for Olkiluoto 1. Three-dimensional **hexagonal dynamics benchmark** (CRE) coordinated, solved and reported, a new benchmark on boron dilution defined. AER **Symposium on VVER Reactor Physics and Reactor Safety**, **Nordic Reactor Physics** (NRF) Meeting and Working Group on VVER safety organized.

Planned results in 1997:

<u>Reactor dynamics</u> **TRAB-3D** core model completed and validated, and supplied with a circuit model (both TRAB and SMABRE available). **HEXTRAN-PLIM** core model updated, first components for the **VVER circuit model** based on PLIM developed. First version of **TRAB-PLIM** circuit validated. **CFDPLIM** improved and tested for modelling of complex thermal hydraulic phenomena. First version of **sub-channel modelling** for hot channel analyses based on PLIM completed. **SFAV** applied to critical flows, countercurrent flows, horizontal flows and flows with greatly varying void fraction.
<u>Reactor physics</u> Improved methods for generating continuous energy and multigroup **nuclear data libraries**. Validation of the hexagonal **CASMO-4**. Updating and validation of **criticality safety methods** (CASMO-4, MCNP, KENO-6). Validation of the SCALE-4.3-program package for **radiation dose calculations and criticality**. Development of **BWR group constants** for TRAB-3D. **Pin power reconstruction** modelling started (HEXBU, HEXTRAN, TRAB-3D). Reactor physics of **future nuclear fuel cycles** (NEA).

VVER safety, international co-operation, benchmark calculations 4th three-dimensional hexagonal dynamics **benchmark**, boron dilution in core, solved, coordinated and reported. Other benchmarks of NEA/NSC, NEA/CSNI and AER solved, if relevant to validation of the calculation system.

Planned later results in 1998 and beyond:

TRAB-3D validated for eg stability calculations. Thermal hydraulics of TRAB-3D modelled with PLIM. Complete VVER circuit model for HEXTRAN based on the PLIM solution method. CFDPLIM improved and tested for optimal application in reactor dynamics codes. Improved hot channel models with sub-channel modelling based on PLIM. First application of the SFAV in dynamics codes, aiming at SFAV-based thermal hydraulics in all reactor dynamics codes. Calculation methods for criticality safety updated and validated. Pin power reconstruction in use with the HEXBU-type nodalization methods (HEXTRAN, TRAB-3D). Improved fuel models in reactor dynamics codes.

Remarks

The work on the SFAV model is funded mainly with a personal grant from the Academy of Finland, from mid-1994 until mid-1997. The project employs one full-time and two half-time research trainees (10 h a week), and is still at present somewhat understaffed. Monte Carlo technique is also applied in the FFUSION research program. The development of the core loading program CORFU is carried out in a separate project funded by VTT.

Applications

A comprehensive and independent reactor physics and dynamics code system has been created at VTT Energy for both BWR and VVER reactors. The code system has been widely used by the nuclear safety authorities and by the Finnish nuclear power companies as well as by customers abroad.

The codes which are presently developed and taken into use will be immediately applied in the safety analyses of Finnish NPPs. The expertise will also be applied in international contract research, as well as for improving safety on VVERs in Central and Eastern Europe.

Thermal Hydraulic Experiments and Analyses (TEKOJA)

Duration	1995 - 1998
Project manager	Dr. Jari Tuunanen
Institute	VTT Energy
Volume and funding 97 - 98	97: 4.4 person y. 1.6 MFIM, 98: 5 person y. 1.9 MFIM
Funding sources in 97 (MFIM)	KTM 1.3 VTT 0.3

Objectives

The general objective of the TEKOJA experiments is to investigate thermal hydraulic phenomena of VVER type pressurized water reactors in accidental situations. The experiments can be divided into two parts. The first part includes experiments for the investigation of basic thermal hydraulics of VVER reactors. The objective of these experiments is to investigate influences of specific features of VVER reactors, such as horizontal steam generators and hot/cold leg loop seals, on the natural circulation characteristics of VVER reactors. The second group of experiments can be called accident management experiments. The objective of these experiments is to investigate possible accident management measures of VVER plants. The third objective of the experiments is to provide data for validation of thermal hydraulic computer codes, such as APROS.

Main results so far

To fulfil the first objective above (basic thermal hydraulics of VVERs) a series of natural circulation experiments with different primary/secondary pressures has been carried out. The experiments widen the data base on VVER natural circulation to low primary/secondary pressures. The experiments showed that pressure is an important parameter on natural circulation. Pressure has influence on the mass flow rate characteristics and transition position of different natural circulation modes. For example, pressure influences on the level swell in the core and, thus, on the transition from two phase to boiler condenser mode natural circulation. Pressure has also influence on the flow regimes and primary pressure behaviour during transition from single to two phase flow.

Concerning the second objective above (accident management studies) SBLOCA and primary secondary leakage (PRISE) experiments have been carried out. In these experiments accident management procedures of Loviisa plant were followed. The experiments focused on the possibility of forming of a non-borated water plug into the primary circuit. In SBLOCA this is possible during boiler condenser mode natural circulation phase. In the PRISE situations non-borated water can flow from secondary to primary side if primary pressure drops below secondary side pressure. In SBLOCA situations forming of non-borated water plug into primary circuit is possible, but the situation is very sensitive to the break size. In PRISE situations flow reversal in the steam generator, i.e. flow of non-borated water from secondary to primary side, was clear in experiments where primary feed and bleed was used as accident management measure.

Planned results in 1997:

The main objective of the year 1997 is to start experimentation in the area of ATWS situations. ATWS experiments will be started with two basic experimental series. The first series investigates maximum pressure scenario in control rod withdrawal situation. The second series investigates start-up of two phase natural circulation from boiler condenser mode by charging pressurizer water.

The first series is planed to fulfil the third objective above (provide data for code validation). The second series covers the first and second objectives above.

Planned results in 1998:

ATWS experiments will be continued in 1998 with a series of experiments with feedback from core temperature and void fraction to core power. The second series in 1998 investigates influences of non condensable gas on natural circulation. In this series experiments with different natural circulation modes and noncondensable gas concentrations will be carried out. The third series will include experiments with steam generator collector rupture. The first and third series would require modifications to the process control system of PACTEL.

Applications

TEKOJA experiments produce data for validation of thermal hydraulic computer codes, support development of accident management measures and provide information about safety system performance in accidental situations. Experimental data is used as Finnish contribution to international programs, such as CAMP or UPTF/TRAM. The experimental data can also be used as a Finnish contribution to programs to improve nuclear safety in Eastern European countries.

Passive Safety Injection Experiments (PAHKO)

Duration	1995 - 1998
Project manager	Dr. Jari Tuunanen
Institute	VTT Energy
Volume and funding 97 - 98	97: 3.1 person y. 1.37 MFIM, 98: 1.5 person y. 0.85 MFIM
Funding sources in 97 (MFIM)	EC 0,35 LTKK 0,29 VTT 0,73

Objectives
The objective of the PAHKO experiments is to study performance of a passive safety injection system in small break loss of coolant accident situations. The passive safety injection system investigated consists of a passive safety injection tank called CMT and two pipelines (pressure balancing line connecting the tank to one cold leg, injection line connecting the tank to downcomer). The second objective of the project is to simulate experiments with APROS code to identify possible modelling deficiencies of the code in the simulation of passive safety injection systems.

Main results so far
The first experimental series of PAHKO experiments have been carried out and analysed. No problems with rapid condensation was observed, such as observed in earlier passive safety injection experiments in PACTEL. The experiments support application of McAdams correlation for calculation of heat transfer from hot liquid layer to the CMT walls, as proposed by Westinghouse. Nusselt's film condensation correlation can also been suggested for calculation of condensation to the CMT walls. The main interest in the following experiments is in very small break situations, where recirculation phase of CMT operation is long. If the recirculation phase is long the whole CMT can come full of water before CMT injection starts. This might cause problems for start-up of safety injection.
Simulation of the experiment GDE-24 of the first series has been started with APROS code. The base case calculation is ready and the documentation and parameter variations have been started. The main problem of simulations is the oscillations of safety injection mass flow rate from the CMT. This is possibly caused by overestimated condensation in the CMT.

Planned results in 1997:
The main objective of the year 1997 is to analyse the second and run the third experimental series of PAHKO experiments, as planned in the contract with the European Commission. The second series investigates the influences of break location on passive safety injection system performance. The main interest in the second series is in very small breaks, where recirculation phase is long. In the third series the CMT will be moved to higher elevation, to increase driving head for safety injection. The series will also include investigations of influences of operator actions on CMT behaviour. The second objective of the year 1997 is to continue analyses of the experiments with APROS code.

Planned results in 1998:
The main objective of year 1998 is to analyse and document the third series of PAHKO experiments, and to write the final report of the project according to the contract with the Commission. The second objective is to analyse experiments from third series with APROS code, and to write conclusions of APROS code applicability on the simulation of passive safety injection systems.

Applications

PAHKO experiments produce data for validation and improvements of thermal hydraulic computer codes and provide new information about passive safety injection system performance in accidental situations.

Reactor Accidents' Phenomena and Simulation (ROIMA)

Duration	1997 - 1999
Project manager	Mr. Risto Sairanen
Institute	VTT Energy
Volume and funding 97 - 98	97: 8.1 person y. 6.385 MFIM, 98: 8 person y. 6.2 MFIM
Funding sources in 97 (MFIM)	KTM 1.700 STUK 0.155 IVO 0.615 TVO 0.360 EU 1.350 NKS 0.165 VTT 2.040

Objectives

The ROIMA-project will be started at the beginning of the year 1997 and planned for three years. The general objectives of the project are:

- Develop and validate the calculation tools needed to plan preventive measures and to train the personnel to severe accident mitigation. A central tool is the APROS process simulator. Thermal hydraulic validation of the code will be performed. Documentation of the code in nuclear applications will be updated. APROS will be extended with selected severe accident models to provide a tool for severe accident training.

- Investigate behaviour of fission products in severe accidents concentrating on combining own experimental work with the research done in EC. The objectives and phenomena selected include fission product chemistry in primary circuit, revaporization and fission product behaviour in containment.
- To reduce uncertainties in phenomena important for the severe accident plans of the Finnish nuclear power plants. These include melt coolability, recriticality and melt behaviour in pressure vessel lower head.
- To study severe accident management actions, focusing on pressure vessel failure mode and long term (up to 1 year) accident management.

Main results so far

The project continues the work performed at the Severe Accident Management (VAHTI) project. The accomplishments of the VAHTI project include:
Validity of the APROS thermal hydraulics has been extended to shutdown state conditions by participation to OECD international standard problem ISP 38. APROS containment model has been improved suitable for describing severe accident phenomena.
Aerosol experiments have been conducted with the AHMED test vessel. The AHMED tests have been successful and the data has been provided to developers of aerosol models for model improvement. A CSNI calculation exercise has been arranged based on the AHMED tests.
Cooling of a partially degraded core and recriticality aspects has been studied as a joint Nordic NKS/RAK-2 co-operation.

A computer code PASULA has been developed for analysis of core melt-pressure vessel interactions. The code has been applied to penetrations analysis.

Planned results in 1997:

Validation of the APROS thermal hydraulic models will continue by selected separate effects tests. Validation against integral tests will focus on experiments performed with the PACTEL-facility. The existing APROS containment models will be documented. APROS severe accident models will be extended to in-vessel phenomena by adding material oxidation and melt relocation. For the containment, recombinator and igniter models will be added.

The construction of the fission product revaporisation facility will be completed and tests using radiotracers and chemical simulants are to be conducted. Samples from PHEBUS FPT1 test will also be revaporised in collaboration with the Institute for Transuranium Elements. Two aerosol tests in VICTORIA facility will be conducted and reported to CEC. The project “Primary system chemistry” will be started and first experiments where the nucleation of CsOH/CsI is studied will be conducted. In connection to this project, experiences from the PHEBUS tests will be utilised. Relevance PHEBUS results to Finnish plant conditions will be assessed.

Melt coolability will be studied by assessing applicability of the MACE test results to Finnish plants. Possibilities for own small scale experiments will be investigated. BWR recriticality aspects will be studied in an EC-project in co-operation with Studsvik, SKI and Riso. The NKS/RAK-2 project will be concluded by estimating melt behavior in RPV lower head.

Pressure vessel interaction with core melt will be estimated with the PASULA code. The work will be accomplished in a KTH coordinated EC project, in which the KTH experiments will provide the thermal hydraulic boundary conditions. A preliminary study of important aspects for long term severe accident management.

Planned results in 1998:

Comprehensive validation of the APROS thermal hydraulic models and selection of suitable test cases to be included in the QA routines of the code maintenance. Completion of APROS containment models including severe accidents. Inclusion in APROS melt behaviour in RPV bottom, RPV external cooling and melt behaviour in cavity.

The results from 3 CEC projects on fission product behaviour will be reported and code validation with gained experimental results is started.

The EC recriticality and pressure vessel interaction studies will be completed. The expected outcome at that time will be an estimate of the magnitude of consequences of the recriticality and a capability to estimate the pressure vessel failure mode in most LWR pressure vessel geometries. The long term accident management research will be extended based on the scoping study performed in 1997.

Applications

As validated the APROS code provides a flexible tool to study accident prevention strategies for Finnish and foreign plants, especially VVERs. The severe accident training features will be applied, when the plant modifications have been completed. The aerosol studies provide more accurate estimates of the radioactive source term to environment in accident situations. Investigation of BWR recriticality, pressure vessel failure mode and melt coolability reduce uncertainties in severe accident management actions. Long term accident management has not yet been studied in detail. The actions may have a decisive role in retaining the fission products within the containment up to several years.

Reliability and Risk Analyses (LURI)

Duration	1995 - 1998
Project manager	Mr. Pekka Pyy
Institute	VTT Automation
Volume and funding 97 - 98	97: 2.4 person y. 1.425 MFIM, 98: 2.4 person y. 1.4 MFIM
Funding sources in 97 (MFIM)	KTM 0.430, VTT 0.200, STUK 0.385, NKS 0.210, other 0.200

Objectives

The main objectives of the project are to develop probabilistic methods for nuclear safety related decisions, for modelling complex phenomena and event sequences, to study effects of maintenance into NPP safety and to study more effective methods for the assessment of human reliability and safety critical organisations. Hereby, the project supports probabilistic safety analysis (PSA) and feasible nuclear regulation and safety management processes.

Main results so far

Uses of decision analysis in international perspective have been studied and reported. An application has been carried out for a nuclear power plant and in two different decision situations. A stochastic optimisation model has been created for time dependent processes and applied to the lifetime optimisation of an NPP. A method to integrate reliability and psychological approaches to dynamic decision making situation has been developed in co-operation with ORINT-project. Methods have been developed for the use of expert judgement and uncertainty analysis with case applications. The first part of the study on human errors in NPP maintenance based on fault history data has been completed (draft report exists). A concept of safety indicator system for nuclear power plants has been published. The project has generated a remarkable number of research publications, conference presentations and scientific articles when compared to its small size.

Planned results in 1997:

DECISION SUPPORT AND RISK BASED REGULATION METHODOLOGY

Development and application of decision analytic and statistical tools in dealing with risk

- An interim report on decision theoretical bases for risk regulation
- A doctoral thesis
- Organization of an international seminar on decision analysis methods and applications in May 1997
- Application study to risk based regulation and risk monitoring

Use of expert judgement methodology in modelling uncertain phenomena (weather risk, PSA level 2 etc.)

- Completion of remaining EU Benchmark exercise tasks
- Application to a specific problem (to be defined later on with STUK)
- Reported applications and a scientific publication on the mentioned area manifesting a feasible and a scientifically sound way to utilize expert opinion in quantitative safety studies

Integrated sequence analysis methodology (RAK-1.3)

- Second reported application,
- Contribution to NKS/RAK -1.3 final reporting
- A scientific article on method development and case studies (with ORINT)

ANALYSES OF SAFETY CRITICAL ORGANISATIONS

Study of human errors originating in maintenance (based on plant failure reporting data)

- Study continuation on another Finnish NPP and reporting its results
- Contribution to NKS/RAK-1 final report (RAK-1.4)
- Reporting to IAEA human error data collection programme and to OECD/NEA/CSNI/PWG1
- A scientific article on the safety impact on the maintenance action reliability

Development of HRA for outage conditions (HRAS)
- Continuation of the work begun in 1996
- Application of the developed method to other shutdown cases and contexts
- Reporting (also to IAEA human error data collection programme)
- A scientific article

Study of safety impacts of plant modifications (RAK-1.5)
- Finalizing study on modification work practices
- Reporting, contribution to NKS/RAK-1.1 final report

Assessment of safety work in the Nordic countries (RAK-1.1)
- Safety activity map completion with case studies
- Final report, contribution to NKS/RAK-1.5 final report (RAK-1.1).

Planned results in 1998:

DECISION SUPPORT AND RISK BASED REGULATION METHODOLOGY

- Development and application of decision analytic and statistical tools in dealing with risk (risk based regulation and risk monitoring) - a final report on decision theoretical bases for risk regulation, potentially a specification of a computer tool / tools
- Use of expert judgement methodology, active role in the preparation of the next NKS programme including expert judgment
- Integrated sequence analysis methodology (RAK-1.3) - participation in potential international follow-up projects (EU concerted action)

ANALYSES OF SAFETY CRITICAL ORGANIZATIONS
- Development of assessment techniques of safety critical organizations and their safety culture
- Continuation of HRA development
- Preparation of initiatives for the next NKS programme and projects for the EU V framework programme

Applications

The results of LURI are directly applicable in Finland and they have been reported through many international co-operation projects, e.g. NKS/RAK-1 programme, OECD/CSNI/NEA/PWG 1 and 5 co-operation and IAEA human error data collection research programme. The project contributes to two EU concerted actions.
By using decision analytical methods, decisions will be based more on rational criteria and the involved uncertainties are explicitly expressed, which will improve the quality of decisions. Due to the complexity of decision situations, also advanced tools for incorporating expert judgement in the decision process must be considered. The application areas touched upon LURI are plant lifetime optimisation, testing and maintenance policy optimisation and methodological improvements.
The topics will be highlighted through the organisation of a Seminar on ‘Decision Analysis and its Applications in Safety and Reliability with the European Safety, Reliability and Data Association (ESReDA) in May 1997 in Helsinki.
By assessing safety critical organisations, their practices and human reliability in individual tasks, better view on the whole spectrum of potential risks is generated. Results directly applicable to improved NPP practices and regulation are generated, e.g., in the studies of plant modifications and human errors in maintenance. In other subprojects, alternative models for important areas such as HRA and assessment of safety work are presented.

Human Factors in NPP Operations (ORINT)

Duration	1995 - 1998
Project manager	Dr. Leena Norros
Institute	VTT Automation
Volume and funding 97 - 98	97: 1.2 person y. 0.850 MFIM, 98: 1.2 person y. 0.8 MFIM
Funding sources in 97 (MFIM)	KTM 0.420, VTT 0.200, STUK 0.180, NKS 0.050

Objectives

The aim is to enhance safe operation of power plants through development of human competencies in the control of complex environments. The competencies are studied from the decision making point of view. The methods developed for the analysis of decision making are further modified to provide practical tools for training and continuous learning, design and validation of control room design and control of safety.

Main results during 1995-1996

Development of a new contextual method for the analysis of operators' decision making in natural or simulated settings. The method was named Analysis of Way of Acting (AWA), and a practical application of the method has been developed for TVO Olkiluoto Power Plant to be used as a tool in simulator training.
Concept for the validation of information presentation. The concept includes analysis of information demands and supply, evaluation of the operators' utilization of information during task performance, and evaluation of the appropriateness of the information presentation.

Concept for a contextual analysis of system safety. In the proposed methodology the stochastic and psychological approaches to risk analysis are integrated with the help of common analysis of decision making context.

Planned results in 1997:

The work is organized in three main topics which run through the whole planning period.

1. DEVELOPMENT OF THE APPROPRIATENESS OF CONTROL ROOM INFORMATION

The structure of the validation concept is available. This concept should be utilized in a concrete validation task and developed further.

1.1 Development of a taxonomy of process control situations

The taxonomy is an essential part of the contextual research approach under development at VTT Automation. The taxonomy is needed in the selection of situations for empirical studies, and it also contributes to the evaluation of the validity and generality of the results achieved in studies based on these particular situations. Furthermore, it is useful in the selection and design of scenarios for simulator training programs

The task involves conceptual work with the aim of describing critical features of process control situations. These features define the task demands that situations set for the operators, whose goal is an efficient and safe control of the process. The task includes the following subtasks:

- Comparative analysis of the situation models from own previous work
- Review of relevant literature
- Conceptualization of critical features of situations
- Report

The results of this task contribute to all ongoing research topics, i.e. control room design, development of operators' competencies and control of safety.

1.2 Development of a method for the evaluation of the availability and distribution of process information

As a first phase in the development of the method an analysis of operators' conceptions of the usability of control room information is carried out. The task includes the following subtasks:

- Selection of process situations for the study (notice interaction with 1.1)
- Interviews with the operators concerning their ways of operating the plant in the defined-situation with special emphasis on the availability and distribution of process information
- Analysis of the interview data
- Report

A research proposal has been delivered for STUK concerning this task, which is planned to be carried out in co-operation with TVO Olkiluoto NPP.

2. ANALYSIS OF CONTROL ROOM OPERATORS' COMPETENCIES

2.1 Validation of the AWA-method (Analysis of Way of Acting)

The AWA-method is going to be used in two new experimental studies. These provide simulator data from 5 different scenarios. Measurement characteristics of the AWA-method will be studied from several points of view. Thus the following tasks are going to be carried out:

- Crews' opinions about the method (questionnaire)
- Comparisons of experimental test results:
  - between scenarios: comparisons of the crews' ways of acting in 5 differ ent scenarios
  - between crews: comparison between 6 crews in the same 4 scenarios
  - within crews: comparison of within 11 crews in 2 scenarios
- Conclusions concerning the measurement characteristics of the AWA-method

2.2 Mobilization of subjective resources during emergency situations

The evaluation of the way of acting has so far concentrated on the operators' conceptions of the process (cognitive point of view) and the team performance (the co-operative point of view). In order to increase the realism of the research approach and the validity of the results, it is necessary that performance is also viewed from the energetic point of view. Therefore we are going to analyze the operators' ways of mobilizing and controlling their subjective resources under time constraints and stressful situations. This new topic will first include following tasks:

- Conceptual analysis of the subject matter on the basis of current literature (literature review)
- Eventual focusing on two research questions:
  - the role of metacognitive processes in the selection of situatively relevant ways of acting (literature review)
  - co-operation as a means to cope with stressful situations (literature review)
- Pre-analysis of experimental results including
- Conclusions regarding future work in the area

3. DEVELOPMENT OF AN INTEGRATED APPROACH TO SYSTEM SAFETY

A first concept of a new approach to system safety has been proposed. This concept, including psychological and a stochastic modeling of decision making in accident situations will be applied to previously collected comprehensive simulator data including 11 crews' performance in a particular disturbance situations. The tasks includes the following subtasks:

- Elaboration of the stochastic model on the basis of performance data
- Analysis of the effect of operators' ways of acting on risk
- Report

This task is carried out in co-operation with the LURI-project and will partly be financed by the NKS/RAK 1.3 project.

Anticipated activities in 1998:

1. DEVELOPMENT OF USABILITY OF CONTROL ROOM INFORMATION

The structure of the validation concept is available. This concept should be utilized in a concrete validation task and developed further.

1.1 Evaluation of the new control room concepts in simulator experiments (1998)

2. ANALYSIS OF CONTROL ROOM OPERATORS' COMPETENCIES

2.1 Validation of the AWA-method (Analysis of Way of Acting) as a training tool

- reporting of results

2.2 Mobilization of subjective resources during emergency situations

- analysis of empirical data on the basis of enlarged AWA-approach

3. DEVELOPMENT OF AN INTEGRATED APPROACH TO SYSTEM SAFETY

3.1 Reporting the results of the empirical study and development of the integrated model.

Applications

The ORINT-research project is planned to provide coherent view to problems of on-line decision making in complex dynamic situations. As the result of the three subprojects a comprehensive methodology will be achieved which provides information that can be taken into account in the development of practices in operating nuclear power plants. It is evident that the research may be useful in at least three major application areas, which are control room design, training and development of competencies, and control of safety.

Co-operation

The ORINT-research team has established divergent multidisciplinary research relationships within the NPP field including international connections. The ORINT- research group carries out parallel studies in other process control domains. This cross-domain approach has been found to be generally beneficial.

Administration and Information Activities of the Research Programme (REHTI)

Duration	1995 - 1998
Project manager	Dr. Timo Vanttola
Institute	VTT Energy
Volume and funding 97 - 98	97: 0.5 person y. 0.370 MFIM, 98: 0.7 person y. 0.42 MFIM
Funding sources in 97 (MFIM)	KTM 0.350, EU 0.020

Objectives

The research projects are supervised to fulfil the goals of the research programme as a whole and to keep their scheduled budgets. The administration project also directs and partly takes care of the external information activities of the research programme. The manager of the programme arranges four steering group meetings yearly to report about progress of the research projects. The steering group treats the annual reports and the yearly plans of the programme before they are submitted to the Ministry of Trade and Industry, and prepares necessary modifications to the programme. The manager of the programme also participates the meetings of the technical advisory groups of the research projects.

Main results so far

Progress of the programme has been reported to the steering group three times a year and in each project about as frequently to the corresponding advisory group. The annual research plans and the first annual report have been prepared in co-operation with the steering and advisory groups.

The first internal evaluation about the startup and progress of the programme was arranged among the steering and the advisory groups, and also within the research staff, in the beginning of 1996. The evaluations were generally positive, but the sufficiency of resources to cover the whole research field was questioned.

The final report of the previous research programme on Nuclear Power Plant systems Behaviour and Operational Aspects of Safety YKÄ (1990-1994) was edited and distributed to domestic and foreign specialists. The general plan of the current Reactor Safety Research Programme RETU (1995-1998) was prepared and delivered for domestic specialists. A brochure of the programme was made, as well as corresponding www-pages (http://www.vtt.fi/ene/eneydi/RETU/). Particularly, seminars on ”NPP safety and decision making” and ”Accident analysis methods”. were arranged in 1996. An Interim Report of the programme is under preparation.

Planned results in 1997:

The Interim Report is distributed for domestic and foreign specialists and bodies. The annual report 1996 is prepared and delivered. The progress of the programme is reported for the steering group and the reference groups in 2-4 annual meetings. The research plan for the year 1998 is prepared. Depending on the decision of MTI, an external evaluation of the programme will be arranged in the later part of the year 1997, partly to support research planning after the RETU programme. Coordination and preparation of material from Finland to the Joint Safety Research Index, which is a concerted action in the EU NFS-2 programme.

Planned results in 1998:

Annual report 1997. Normal steering and reference group meetings. Updating of the Joint Safety Research Index. Preparation of the Final Report of the research programme. Final seminars. Final review of the programme (possibly externally organised). Annual report 1998 (completed in the beginning of 1999).

Applications

Discussions between the research bodies and end users of the research results in the steering and the reference groups, together with the summary information about all the research fields of the programme enable identification of weight areas and future resource needs, which allows efficient guidance of the research.