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1995–1998
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Oy Edita Ab, Espoo 1998

Preface

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 programmes, *Reactor Safety (RETU*) and *Structural Integrity of Nuclear Power Plants (RATU2)*, have continued research in most areas of the earlier programmes on *Nuclear Power Plant Systems Behaviour and Operational Aspects of Safety (YKÄ)* and *Nuclear Power Plant Structural Safety (RATU)*. This final symposium report gives an overview of the scientific and technical results obtained during the RETU programme.

The Ministry of Trade and Industry contracted the Technical Research Centre of Finland (VTT) to co-ordinate the programme. VTT also performed most of the research. The funding sources were KTM, VTT, the Radiation and Nuclear Safety Authority (STUK), the Lappeenranta University of Technology, the nuclear power companies Imatran Voima Ltd (IVO) and Teollisuuden Voima Ltd (TVO) and the European Union through its Nuclear Fission Safety Programme NFS-2.

In addition to the editor, several people have contributed to the preparation of this report, in particular the project managers Mr. Seppo Kelppe[1], Ms. Hanna Räty[1], Dr. Jari Tuunanen[1], Mr. Jyrki Kouhia[1], Mr. Risto Sairanen[1], Mr. Pekka Pyy[2] and Dr. Leena Norros[2].

The editor is grateful to all the colleagues participating the work of the programme and assisting in the preparation of this report and the final symposium, as well as to the members of the steering and reference groups for providing their experience in the planning and steering of the research.

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[1] VTT Energy
[2] VTT Automation

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RETU
The Finnish research programme on reactor safety (1995 - 1998)

Timo Vanttola
VTT Energy
Espoo, Finland

1. Introduction

The publication constitutes the final report of the Finnish national research programme on *Reactor Safety (RETU)* conducted in 1995 - 1998. The programme has focused on operational aspects of nuclear safety, such as search of safe limits of nuclear fuel and the reactor core, accident management methods and risk management of the nuclear power plants. This introductory article first describes the general needs, organisation and volume of reactor safety research in the country and then gives an extended summary of the RETU programme.

The body of the publication consists of the articles presented in the final symposium of the programme. They describe the selected main results obtained. The report is supplemented with annexes of international co-operation, publications, list of persons in the programme management and staff, academic degrees awarded and a brief summary of the follow-on programme.

1.1 Nuclear safety in Finland

Finland produces nearly one third of its electricity with two nuclear power plants that are the two VVER-440 units at Loviisa, operated by Imatran Voima Ltd (IVO), and the two BWR units of ABB type at Olkiluoto, operated by Teollisuuden Voima Ltd (TVO). The utilities are responsible for the ultimate nuclear safety of the power plants.

The Finnish Nuclear Energy Law (Ydinenergialaki 1987) requires that “The utilisation of nuclear energy shall be safe, and shall not cause damage to the public, environment, or property". General safety regulations have most recently been determined in a Decision of the Council of State /VNP 395/91) issued in 1991. Supervision of the fulfilment of nuclear safety goals is the duty of the Radiation and Nuclear Safety Authority (STUK) that stipulates the more detailed regulatory guides. These YVL guides, that principally concern new plants, are updated periodically. The safety level of the operating plants is also assessed regularly, and the goal through plant upgrades is to reach a safety level consistent with the new plant requirements. The effort to confirm high level of nuclear safety under changing conditions set by emerging technology, plant upgrades and ageing calls for continuous investment in research, both in the public and in the utility sector.

1.2 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 co-ordinated research programmes. The programmes are organised in such a way that they also support the basic and advanced education of experts and facilitate systematic, efficient international co-operation. The three research areas and the programmes recently completed or under way are:

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

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

Waste management: 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 most of its nuclear energy research budget directly on these three programmes. Other major contributors are the Technical Research Centre of Finland (VTT) and the Radiation and Nuclear Safety Authority (STUK). The power companies, Imatran Voima Ltd (IVO) and Teollisuuden Voima Ltd (TVO) also participate in the funding of many of the projects. In addition to the above programs, there are parallel research programmes on advanced nuclear power plant concepts (ATWS) and on nuclear fusion (FFUSION) in progress, funded mainly by the Technology Development Centre (TEKES), Finland and co-ordinated 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.

1.3 Organisation of the research activities

In Finland nuclear energy research and development has been decentralised 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.

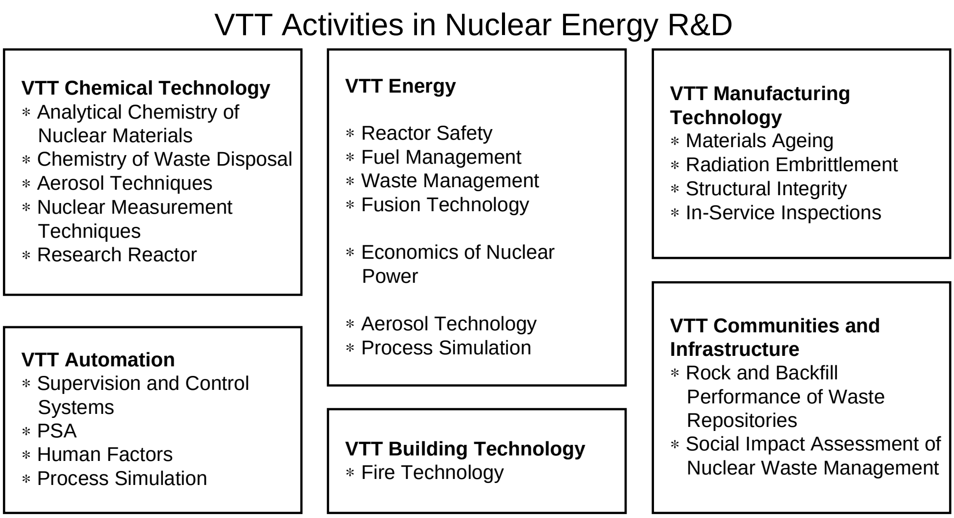

Figure 1. VTT activities in nuclear energy R&D.

Extensive world wide 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.

1.4 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, particularly on the power company side because of the intensive modernisation projects of the Finnish nuclear power plants at that time. About two thirds of the activities were funded by the utilities. The national research programmes cover about 1/4 of the total research volume. 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 includes plant-specific research on reactor materials, severe accident management and thermal-hydraulics, carried out mostly by the utilities. The internal research conducted by STUK mainly focuses on environmental impacts and radiation protection.

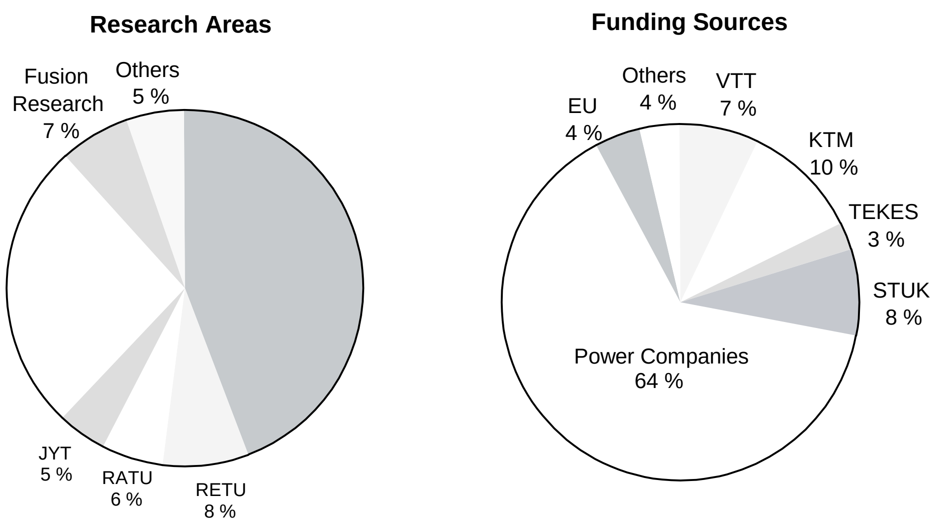

Figure 2. Direction and funding distribution of nuclear energy research in Finland in 1996. The total annual funding is about FIM 180 Million and the research volume is some 200 person years. Research area "Others" includes environmental impacts, research reactor and radiation protection.

2. Outlines of the RETU programme

Duration:	1995-1998
Total funding:	FIM 58 mill.
KTM funding:	FIM 20 mill.
Volume in 1998:	FIM 14.3 mill. 23 person years
Funding in 1998:	
KTM	FIM 4.8 mill.
VTT	FIM 4.9 mill.
STUK	FIM 1.0 mill.
Utilities	FIM 1.7 mill.
EU/NFS-2	FIM 1.3 mill.
Others	FIM 0.6 mill.
Research institutions:	
VTT	
Lappeenranta Univ. Tech. (LTKK)	
Imatran Voima Ltd (IVO)	
Programme management:	
VTT Energy	
Manager: Dr. Timo Vanttola	

The Finnish national research programme on *Reactor Safety (RETU, 1995-1998)* concentrated on 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 organised by KTM.

The programme was supervised by the steering group that represented the main funding sources. Reference groups were set up to support research projects and to promote exchange of information with the end users.

The research fields and their interconnections are presented in Fig. 3. Table 1 lists the projects and provides some overall information.

General objectives of the programme may be listed as follows:

- 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

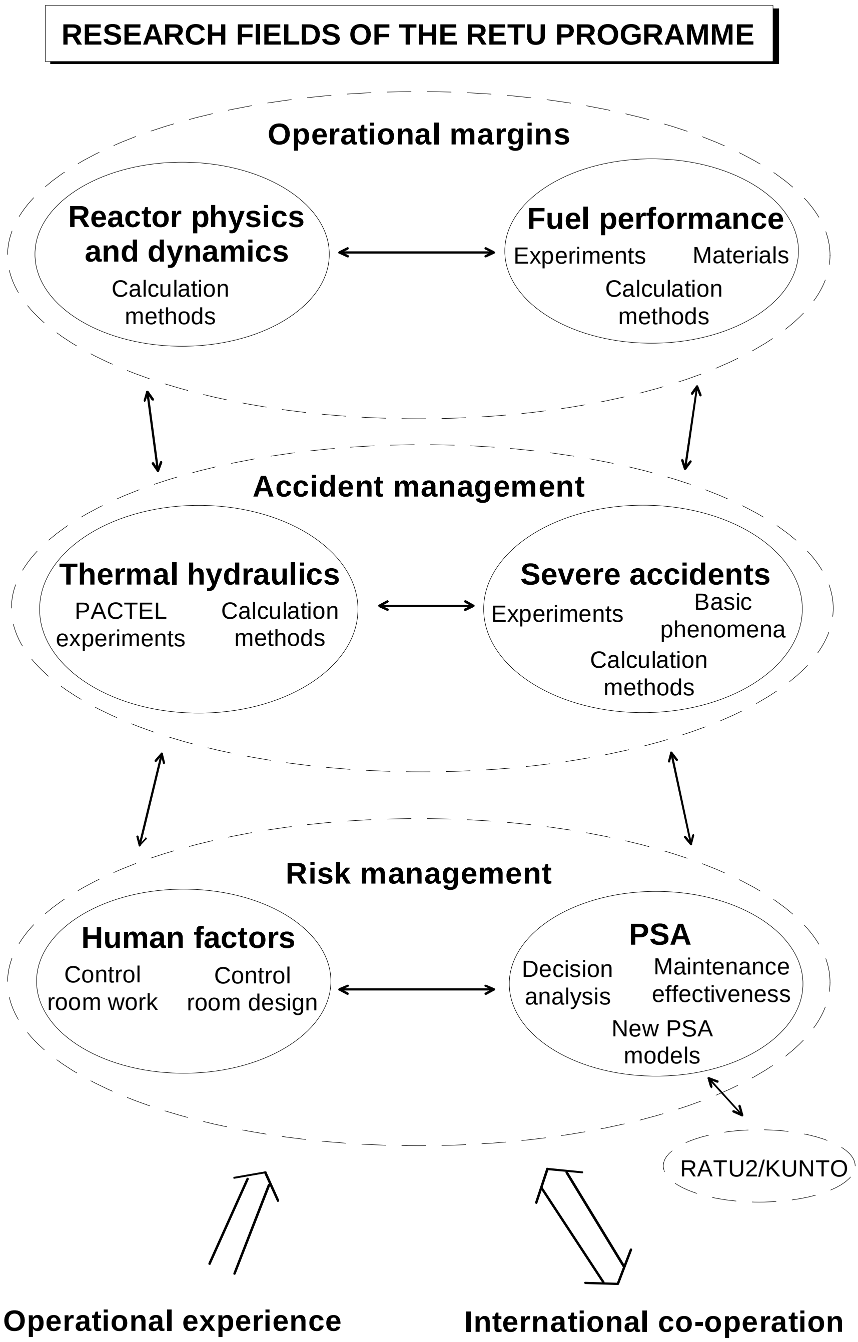

Figure 3. The research fields of the RETU research programme.

Table 1. Projects of the RETU research programme.

Research field Project (Acronym)	Project manager	Duration (years)	Institutions	Volume (person years) -95	-96	-97	-98est.
Operational margins of nuclear fuel and reactor core:							
Transient models of nuclear fuel (PATRA)	Mr. Kari Ranta-Puska (Mr. S. Kelppe -95)	94 –97	VTT Energy	1.7	1.3	1.5	-
Transient behaviour of high burnup fuel (KOTRA)	Mr. Seppo Kelppe	98	VTT Energy	-	-	-	1.7
VVER fuel experiments (SOFIT)	Mr. Risto Teräsvirta	85-95	IVO PE	1.0	-	-	-
Calculation methods of reactor physics and dynamics (DYNAMIC)	Ms. Hanna Räty	95 - 98	VTT Energy	4.1	5.9	5.2	6.7
Assessment of complex transient and accident situations / Accident management:							
Thermal hydraulic experiments and analyses (TEKOJA)	Mr. Jyrki Kouhia (Dr. Jari Tuunanen - 95-97)	95 - 98	VTT Energy and LTKK	8.6	4.9	4.6	4.8
Passive safety injection experiments (PAHKO) (Extracted from the programme in 1998)	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 - 98	VTT Energy and IVO	-	-	8.1	7.5
Management of nuclear power plant risks:							
Reliability and risk analyses (LURI)	Mr. Pekka Pyy	95-98	VTT Automation	2.0	2.3	2.4	1.5
Human factors in NPP operations (ORINT)	Dr. Leena Norros	95-98	VTT Automation	1.1	0.8	1.2	1.1
Administration and information activities (REHTI)	Dr. Timo Vanttola	95-98	VTT Energy	0.7	0.5	0.5	0.7
			Total:	26.0	26.2	26.8	24.0

3. Summary of the main achievements

The objectives and the main results of the research projects in the various fields of the programme are summarised below. The information and education activities are presented shortly at the end of this chapter.

3.1 Nuclear fuel behaviour

The research has been carried out in the PATRA, SOFIT and KOTRA projects.

Objectives

The performance of high burnup fuel may significantly differ from that of fresh fuel. The difference becomes obvious in extreme conditions, like in a postulated Reactivity Initiated Accident (RIA). Recent experience indicates relationship between low enthalpy fuel rod failures and large fission gas inventory in the fuel (rim) and clad hydriding. Better understanding of the influence of various fuel rod and reactor specific parameters on the fuel performance in accident conditions is required. For more accurate evaluations, transient and steady state calculation tools need to be qualified for high burnup and for the fuel types used. This will support the utilities in their reactor modernisation programmes and the authority in setting the fuel and reactor safety criteria.

The objectives of the *Transient models of nuclear fuel* and *Transient Behaviour of High Burnup Fuel* projects (PATRA and KOTRA) have been to acquire the knowledge essential for assuring the safety of high burnup fuel in steady state and transient conditions, and to update the fuel performance models.

Results

<u>Performance and modelling in accidents</u>

An agreement on co-operation in the field of nuclear safety between the French IPSN (Institute de Protection et de Sûreté Nucléaire) and VTT has granted VTT access to IPSN RIA code SCANAIR. VTT has participated in the validation of the code, partly in the SCANAIR development team in the Cadarache Centre. 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. An exercise shows encouraging results for cladding creep in comparison with measurement, while high degree of detail is required to adequately describe the mechanics.

Normal operation and probabilistic assessments

Confidence on the capabilities of the ENIGMA steady state fuel performance code was strengthened by the international code benchmarking programme FUMEX organised by IAEA. As with most of the codes, however, the mechanical interaction could not be predicted well enough. A statistical procedure has been designed and implemented in the ENIGMA code. Making use of this feature, it became possible to apply probabilistic variations of input data and arrive at core-wide distributions of the most important thermal parameters. In a series of such studies, it could be broadly concluded that the changes in the linear heat generation distributions in the context of the modernisation of the Finnish plants will most probably lead to less demanding thermal conditions for the fuel despite the uprated reactor powers. The performance of the code for high burnup conditions has been improved, whereby data from the OECD Halden Project experiments have been most useful.

VVER test reactor irradiations

The Russian-Finnish *VVER fuel experiments* research programme (SOFIT) has yielded a fair amount of quite unique measurement data. Much of the in-pile data and PIE results were released and submitted to NEA and are now included in an international data base, through which they have been widely used for code validation by several foreign organisations, as well. Due to problems with the MR reactor operation in Moscow, the programme had to be prematurely terminated in 1995.

International research programmes

The Second Rod Over Pressure Experiment (ROPE II) and the Defect Fuel Experiments (DEFEX) hosted by Studsvik Nuclear AB were carried through. Reviews and adaptation of the result to Finnish conditions were prepared.

An internationally sponsored experimental programme is being launched by IPSN at the CABRI Reactor, Cadarache, to resolve the remaining issues of high burnup fuel in accidents, with anticipated Finnish participation.

3.2 Reactor physics and dynamics

The research has been carried out in the DYNAMIC project.

Objectives

A comprehensive and independent reactor physics and dynamics code system has been created at VTT Energy for both BWR and VVER reactors. At present most of the licensing analyses of the Finnish nuclear power plants can be performed with our own codes. The main objective of DYNAMIC project has been 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. The applications of the reactor physics code system include: generation of nuclear data for fuel management codes, reactor core simulators, and transient and accident analysis codes; flux and dose rate calculations (eg. pressure vessel irradiation and other out-of-core calculations) and criticality safety studies; calculation of initial data for studies on final disposal of spent fuel and decommissioning; studies of new fuel types, fuel cycles and fusion neutronics. 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.

Results

Development and validation of three-dimensional dynamics codes

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 hexagonal nodal flux model of the HEXTRAN code has been successfully validated against accurate fine-mesh finite-difference calculations. Excellent agreement with POLCA-4 results has been obtained in stationary full power calculations of the Finnish Olkiluoto BWRs. Calculations of three-dimensional PWR control rod ejection and control group withdrawal benchmarks of NEA generally show a good agreement with the reference results, and the results of a BWR benchmark are compatible with other published results. Coupled with the existing BWR circuit model the code gives essentially similar results to VTT's one-dimensional TRAB code for a real Olkiluoto transient. The validation of the code will continue with comparisons against plant data of startup and transient measurements of the Olkiluoto reactors. Validation with international benchmarks will continue coupled with a PWR circuit model. The main applications of TRAB-3D will be transient and accident analyses (RIA, ATWS, stability, etc) for BWRs.

Improvements in thermal hydraulics modelling

New versions of the one-dimensional TRAB code and the three-dimensional HEXTRAN code utilizing VTT's new hydraulics solution method in the core have been successfully tested in both VVER and BWR conditions. Results for BWR 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. HEXTRAN-PLIM-results for a demanding boron dilution benchmark show the superiority of the new method against conventional or improved boron tracking methods: also the temperature disturbance is calculated accurately. 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.

Updating and validation of reactor physics code system

The reactor physics calculation system has been updated by taking into use many international codes or their latest releases, eg. the latest version of the NJOY nuclear data processing code, the SCALE4.3 program package, the MCNP4B Monte Carlo code, the advanced three-dimensional two-group reactor analysis code SIMULATE-3, and the first production version of the fuel assembly program CASMO-4 (for square, hexagonal and cluster geometry).

The nuclear data processing system has been applied to generate programwise continuous energy and multigroup 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. The criticality safety calculation tools have been validated for VVER applications (eg. storage pool criticality) against experimental data from critical facilities, and calculational burnup credit benchmarks. The hexagonal version of CASMO-4 has been validated with code comparisons, against experimental data from critical facilities and by generating nuclear data for simulations of the Loviisa units with good results. The applicability of the codes and libraries has been studied in order to establish their advantages and weak points. The updating and validation of the code system will be continued with the purpose of ensuring that the code system functions properly and that it can be used correctly.

Contributing to international VVER safety

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 were followed by a benchmark on a reactivity accident initiated by a local boron dilution in the core. During the benchmark calculation improved boron tracking methods were taken into use in all the applied codes, and reasonably good agreement was achieved among the results of all participants from five countries. The presently ongoing benchmark calculation is a steam header break in a VVER reactor, and includes also the cooling circuit models.

Training aspects

The project encourages both graduate and undergraduate studies: two doctoral theses and one masters' thesis have been completed and two master's theses are in progress. Several undergraduate research trainees have been employed annually.

3.3 Thermal hydraulic experiments and analyses for VVER and ALWR plants

Thermal hydraulic experiments and analyses for VVER and ALWR nuclear power plants included two projects, which were TEKOJA project (1995-1998) and PAHKO project (1996-1997). The TEKOJA project concentrated on the thermal hydraulics of VVER type nuclear reactors used in Finland, where as operation of a passive safety injection system was investigated in the PAHKO project. This kind of safety system has been planned for new nuclear reactors.

Objectives

The general objective of the thermal hydraulic experiments was to investigate thermal hydraulic phenomena in nuclear reactors during accidental situations. The experiments were divided into three parts. The first part included tests for the investigation of basic thermal hydraulics of VVER reactors. The objective of these experiments was to study influences of specific features of the VVER reactors, such as horizontal steam generators and hot/cold leg loop seals, on the natural circulation characteristics of the VVER reactors. The second group of experiments was called accident management experiments. The objective of these experiments was to investigate possible accident management measures of the VVER plants. The third set of tests included experiments where the objective was to study performance of a passive safety injection system in small break loss of coolant accident situations. Such a safety injection system has been proposed for a new reactor concept. The passive safety injection system investigated consisted of a passive safety injection tank (called core make-up tank), a pressure balancing line connecting the tank to a cold leg, and an injection line connecting the tank to downcomer. The two first-mentioned categories were included in TEKOJA, while the last group of tests was carried out in the PAHKO project.

The common objective for all of these tests was to provide data for validation of thermal hydraulic computer codes, such as APROS. In the PAHKO project the experiments were simulated with APROS code to identify possible modelling deficiencies of the code in the simulation of passive safety injection systems. In the TEKOJA project the computer codes were used to help the planning of the tests and to assist the analysis of the test results, although the validation of the computer programs as such was outside the scope of the project.

Results

Basic thermal hydraulic experiments of VVER reactors

A series of natural circulation experiments (SIR-20, SIR-21, SIR-22, SIR-23) were ran using the PACTEL test loop. The test results revealed, that pressure has an important effect on the primary loop mass flow rate characteristics in natural circulation. Due to the higher level swell in the upper plenum (and the higher steam velocity in the hot leg) in low pressure, the maximum loop flow is measured in two-phase natural circulation mode, rather than in single-phase natural circulation regime, which is the case in high pressure. This test series indicated also that the transition primary coolant inventory, when the loop flow changes from one natural circulation heat transfer mode to an other, is depending on the pressure.

An other set of experiments (LSR-20, LSR-21, LSR-22) was performed to study the hot leg loop seal effect on natural circulation. These tests demonstrated how cyclic behaviour of loop flow was replaced by a steady two-phase flow, when the primary pressure was reduced.

Two tests were conducted to study the transition from single-phase natural circulation to two-phase natural circulation heat transfer mode (ATWS-1, ATWS-4). The aim in the experiments was to examine if the shift between the two flow regimes is possible without flow stagnation caused by hot leg loop seal. The low primary pressure and high core power delayed the loop flow stagnation, but in PACTEL the flow stopped each time in transition period.

<u>Accident management experiments</u>

Two test series addressed the problem of possible boron dilution. The first group of experiments focused on a primary to secondary leak incident and an effect of operator actions on possible flow reversal in the break, that is flow direction from secondary to primary side (PSL-5, PSL-6, PSL-7). That means that low borated water would flow to the primary loop. A clear flow reversal was observed in the tests where primary bleed was used to cool down the primary side.

Mass of low borated water during boiler-condenser natural circulation flow was determined in a set of small break loss of coolant accident experiments (SBL-30, SBL-31, SBL-32, SBL-33). The mass of condensate determined from the test results was about 35 % higher than in the reference calculation for the power plant.

An examination on the effect of different operator measures in order to force coolant from the pressurizer back to the rest of the primary loop was performed. The study included two experiments (ATWS-2, ATWS-3). Injection of cold water on the top of a uncovered heat exchange tube bundle of a steam generator was observed to be very efficient way of draining the pressurizer.

In an loss of feed water together with failure of scram transient analysis for Loviisa power plant showed how heat transfer regularly shifted from one loop to an other. This behaviour was duplicated in a PACTEL test ATWS-6. The primary reason for the switching of an active steam generator was deteriorating heat transfer in a steam generator when the secondary side level dropped.

Two tests were performed to study pressurizer behaviour during primary loop depressurization (CPR-11, CPR-12). The axial void fraction profile was uniform in these tests unlike in full scale UPTF experiments.

Data for validation and applications

Several tests were conducted to investigate the pressurizer behaviour in ATWS transients. The particular interest in the tests was compression and expansion of a large steam bubble in the top of the pressurizer when coolant in surging in and out of the pressurizer. This data will be used for code validation.

Many of the performed tests has been simulted with computer codes. APROS has been used to analyse small break loss-of-coolant transients. RELAP calculations has performed on depressurization tests, primary to secondary leak experiments and stepwise inventory reduction tests.

Performance of a passive safety injection system in small breaks

Three sets of experiments were performed in the PAHKO project to study the operation of the passive safety injection systems in small break loss of coolant accident. Each set included five tests. The first test series focused on the influence of break size on the safety system performance. The second test series studied the effect of break location on core make-up tank behaviour. The last set of tests investigated the influence of core make-up tank location on the operation of safety injection system.

In the course of the small break loss-of-coolant transient three phases could be distinguished. During the first phase (recirculation), hot water from the cold leg flowed to the core make-up tank through pressure balancing line, while cold liquid from tank flowed to the downcomer through injection line. The next phase started when the level in the cold leg dropped at the elevation of pressure balancing line connection and two-phase mixture flow to the tank (oscillation phase). Mass flow oscillated partly because of condensation of steam at the top of the make-up tank. When cold leg level dropped clearly below the pressure balancing line connection, an injection phase started, where steam replaced the out flowing liquid.

There was no core heat-up during operation of the core make-up tank in any of the tests. A sparger must be installed at the end of the pressure balancing line at the top of the core make-up tank. Without the sparger intense condensation was observed in the safety injection tank. McAdams correlation can be applied to calculate the heat transfer from the hot liquid layer to the tank wall. The analyses of the test results suggested the use Nusselt's film condensation correlation for calculation of condensation to the tank wall.

Validation of codes against passive safety injection experiments

The general behaviour of the calculated transients was well duplicated by APROS. However, the code could not correctly simulate the thermal stratification in the core make-up tank. The deficiency of stratification model in the computer code resulted in oscillations in the safety injection mass flow rate. This suggested that some of the physical models used in safety analysis computer codes are not sufficient for analyses of proposed passive safety systems, without modifications.

3.4 Severe accident research

The research has been carried out in the VAHTI and ROIMA projects.

Objectives

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 is 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 study severe accident management actions, focusing on pressure vessel failure mode and long term (up to 1 year) accident management,

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.

Results

Development and validation of calculation tools

Thermal hydraulic models of the Finnish Advanced Process Simulation Environment, APROS, have been validated and improved. Recently, the code has been tested in the OECD/NEA benchmark on TMI-1 main steam line break exercise. The APROS thermal hydraulic description has been extended suitable to shutdown state analyses. The APROS system has been used to create a plant model of the Olkiluoto BWR. 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 plant model generally calculated the BWR cases well, but improvement needs were identified for some submodels. The VVER-440 model of APROS has been improved in co-operation with IVO. The APROS code has been extensively applied in the current license renewal and power uprating analyses of the Loviisa plant.

The APROS process simulator is being extended to cover selected severe accident phenomena for training purposes. The VAHTI and ROIMA projects have contributed improvements in the APROS containment model to describe the physical conditions and accident management hardware relevant for severe accidents. The containment model has been documented. The phenomena currently modelled 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. The containment model has been tested by calculating a LOCA test performed at the full-scale Marviken suppression pool containment facility. The new APROS containment model could calculate well the general progression of the test. Causes for deviations are mainly due to phenomena that can not be accurately modelled with a lumped parameter description. A set of specification reports have been prepared to determine the properties of the in-vessel and fission product models. A detailed model for high temperature oxidation of Zr-alloys and steel components, and a report of recommended material properties have been completed. The development of reflooding models has been initiated.

Behaviour of fission products in severe accidents

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. The fission product experiments continued in the EU project Aerosol Physics in Containment utilising the VICTORIA containment facility for a total of 9 thermal hydraulic and aerosol tests. The AHMED experiments and three VICTORIA tests have been analysed with the CONTAIN code. The calculations provided necessary validation prior the code application to plant analyses. It was concluded that the code could accepted in the PSA Level-2 analyses of the Loviisa plant. Fission product revaporization under primary circuit conditions has been investigated experimentally with thermal gradient tube facility. Studies on nucleation of CsI/CsOH and reactions of CsI with boric acid and structural materials have been conducted in the EU project Fission product vapour / aerosol chemistry in the primary circuit.

Studies of severe accident management actions

The 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 question is important due to the SAM adopted for the Olkiluoto BWRs, which specifies flooding of the lower drywell prior to the vessel failure. The bottom head failure mode is one of the key parameters to determine the extent of melt-water interaction. 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 analysed. 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. The creep model has been tested by comparing the PASULA calculations with experimental results from the EU Revisa project. 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. The PASULA code has also been applied within the EU Core Melt-Pressure Vessel Interaction (MVI)-project.

Reduction of uncertainties in severe accident phenomena

The applicability of current MACE experiment results for Finnish nuclear power plants has been assessed. The evaluation considers the general observations seen in the MACE tests, scaling aspects, and material differences between the MACE experiments and Finnish plants. Heat transfer characteristics have been derived from the early MACE tests M0 and M1b. The results have been used to produce coolability estimates for the plant specific accident sequences in the Finnish plants. Typically the core melt would cool from the 2500 K initial temperature to the solidus temperature of the silicate concrete, 1400 K, in 2-3 hours both in case of the Olkiluoto and Loviisa plants. The concrete floor would be eroded in this time by about 10 cm.

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 front modelling. Recriticality studies are continued in the Severe Accident Recriticality Analyses (SARA) project, which is a part of the EU Nuclear Fission Safety research programme

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 over predicted at the late phase of the experiment. The deviation was most probable caused by inaccurate modelling of heat losses

3.5 Reliability and risk analysis

The research has been carried out in the LURI project.

Objectives

The main objectives of the project are to develop probability based 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.

Results

<u>Methods for nuclear safety related decisions</u>

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. Their development and uses have be promoted in the LURI project. Uses of decision analysis in nuclear power industries have been mapped internationally. To spread the approaches to practical decision making, example applications have been carried out for a nuclear power plant and for STUK regulatory decision making. A stochastic optimisation model has been created for time dependent processes and applied to the lifetime optimisation of an NPP. A doctoral thesis has been published in the area.

Expert judgement and uncertainty in safety analyses

Uncertainty analysis and expert judgement are an important part of PSA work. Actually, they are used in every PSA, but mostly too much embedded in the models and data. In the LURI project, a method has been reported to express model uncertainty explicitly in safety studies. The work on expert judgement area has been extensive - a method to completely carry through an expert judgement has been reported and tested in many applications. Especially, the method concentrates on unbiased elicitation of judgements and combining them by using Bayesian statistics.

Methods for the assessment of human reliability

Human reliability analysis has always been a problem area of PSA. The exclusion of human factor from PSAs would, however, lead to a biased view on safety level. The LURI project has approached the problem by establishing an integrated co-operation with psychology. One of the conclusions conclusion of this work is that decision analytic and dynamic stochastic process models suit well to detailed modelling of human reliability. A method to integrate reliability and psychological approaches to dynamic decision making situation has been developed and reported. An EU concerted action on integrated methodologies has been initiated to further extend the point of view. In LURI, also methods for commission error studies and for treatment of human actions in level 2 PSA studies have been created.

Effects of maintenance into NPP safety

A study on human errors in NPP maintenance based on fault history data has been carried out. The study is quite unique because a large amount of plant maintenance data has been used and there has been a great deal of communication between the researchers and the plant staff. The study has produced a report that brings new information about human errors related to maintenance, e.g. that instrumentation and control equipment are an important group. A more thorough study has been carried out for dependent errors.

Safety and high reliability organisations

In this area, a comparison of Finnish and Swedish regulatory bodies has been carried out, a conceptual model of NPP safety work has been presented and a model for licensing procedure of plant modifications has been presented. The regulatory body comparison shows that the few differences between the countries are generally related to the different histories of nuclear power in them. The modification study has shown that it is possible and advantageous to use formal modelling to organisational processes, too. This area has contributed to the birth of an EU concerted action on organisational factors (ORFA).

3.6 Human factors

The research has been carried out in the ORINT project.

Objectives

The aim is to enhance safe operation of power plants through development of human competencies in the control of the complex environment. 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.

Results

Control room operators' working practices

With the premise of the intentional nature of human activity we have developed a new approach for the analysis of activity in real-life situations. We call the methodology Contextual Analysis of Working Practices (CAWP). Habit of action is a central concept, and we have proposed a practical way to identify habits of action through the analysis of the actors' ways of taking account of the possibilities and constraints of the situation and using available resources. During the development of the CAWP methodology we have carried out studies in two nuclear power plants and executed four series of simulator experiments. This has taken place in close co-operation with the simulator trainers and experts of the plants, and nearly all control room crews of these plants have been involved.

In the first simulator study NPP operators' working practices were analysed. Two distinct habits of action could be identified. In the first, the interpretative habit of action, the crews seemed to pursue towards an interpretation of the nature of situation with the help of active search for information and shared process dynamic reasoning. Another pattern of interactions, the procedural habit of action, was also identified. Typical to it was that operators used standard methods for stabilisation and did not seem to search for diagnostically critical information. In their mutual communication, they did not express attempts to pursue an interpretation of the particular disturbance. The results completed our assumptions concerning the NPP process operators' core task. Tendency towards a coherent and situation-specific interpretation of the behaviour of the process could be considered decisive for an adaptive interaction with he process. In a later simulator study with 6 crews in four disturbance situations, we assessed the crews' habits of action with a method including 34 concrete items expressing the above evaluation categories. It was found out that habits of action with clear tendency towards coherent and situation-specific interpretation correlated in general with the high adequacy of process control. In particular the chief supervisor's management style explained variance in process control. In this study we also found indications of the significance of working practices for a situationally adaptive use of information aids in the control room, which ought to be verified later.

Development of a simulator training method

In co-operation with the simulator training centre of one NPP we adopted a further aim to design a practical method for the evaluation of the operators' performance during simulator training. Through providing means for the conceptualisation of the crew's decision making in context the developed method brings about the essential features of activity in relation to the demands and boundary conditions set by the situation. It provides concrete, situation specific criteria for the crews' task performance and habits of action. The method is aimed to enhance the trainees' competence. The idea is that by helping to make the trainees' bases of inference more explicit the method promotes reflectivity and learning. The conceptualisation of the task situations does not only provide bases for the self-reflection among the operators but it also invites the instructors to explicate the criteria they consider relevant for learning situations. Therefore, the method also promotes the instructors' work and competencies.

Methodology for the validation of information presentation

We have studied the usability of the control room information systems in two different studies. In the first one a series of simulator experiment was carried out for the validation of a information aid to be used in disturbance situations. We applied the concept of habit of action as a tool for the analysis of the significance of the different information conditions for the management of the process. We also evaluated the effect of the use of the information system on the performance. The criteria for adequacy of process control were comprehensive and context-dependent. In carrying out this task the opportunity was used to develop further our method for the conceptualisation of disturbance situations as decision-making contexts. The aim was to develop a taxonomy for the process control situations. In a further study, focused on the analysis of control room alarm systems, development of evaluation criteria for situational appropriateness of control room information has been carried out.

Integrated approach to system safety

In co-operation with experts of reliability analysis (LURI-project in this program) we have developed a new approach to analyse operators' decision-making and to perform human reliability analysis (HRA) in accident sequences. The approach is based on the integration of two complementary approaches, probabilistic modelling and decision theory, and contextual human factors psychology. In both "sides" we first, adopted methods that adequately reflect essential features of process control activity. Then, we carried out an interactive HRA process. During this process the stochastic marked point process model was used as a tool by which the psychological approach could be interpreted and utilised for reliability analysis. An application of the methodology was demonstrated.

3.7 Information exchange and education of experts

The steering group and the reference groups of the programme met two to three times a year to discuss the achievements, priorities of future topics and annual work plans of the projects. Seminars were arranged annually or biannually for a wider audience in most of the research fields, and the results summarised in this report are presented in the final seminar of the programme. Main international connections appear in Appendix A.

The programme has been reviewed both by the reference groups and by foreign experts (Faidy & Hayns 1998) with generally very good results, and their remarks have been taken into account in the future plans.

The programme has maintained its internet pages in the adress *http://www.vtt.fi-/ene/eneydi/RETU/*. The publications of the research projects appear in Appendix B. An extensive interim report was submitted for international distribution (Vanttola & Puska 1997).

Statistics of publications (see Appendix B):

• Papers in scientific journals	17
• Conference papers	158
• Research institute reports	31
• Others	174

The following numbers of academic degees were awarded based on the work in the projects (see Appendix C):

• Doctor	3
• Licentiate	2
• Master of Science	2

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Faidy, C. & Hayns, M.R. Evaluation of the RATU2 and RETU Research Programs. Ministry of Trade and Industry, Finland, Studies and Reports 8/1998.

Vanttola, T., Puska, E-K. (eds.). RETU. The Finnish Research Programme on Reactor Safety. Interin Report 1995 - May 1997. Espoo 1997, Technical Research Centre of Finland, VTT Research Notes 1856. 168 p. + app. 51 p.

Probabilistic assessments of fuel performance

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Abstract

The probabilistic Monte Carlo Method, coupled with quasi-random sampling, is applied for the fuel performance analyses. By using known distributions of fabrication parameters and real power histories with their randomly selected combinations, and by making a large number of ENIGMA code calculations, one expects to find out the state of the whole reactor fuel. Good statistics requires thousands of runs. A sample case representing VVER-440 reactor fuel indicates relatively low fuel temperatures and mainly athermal fission gas release if any. The rod internal pressure remains typically below 2.5 MPa, which leaves a large margin to the system pressure of 12 MPa. Gap conductance, an essential parameter in the accident evaluations, shows no decrease from its start-of-life value.

1. The ENIGMA Fuel Performance Code

For steady state normal conditions, VTT Energy runs the originally British ENIGMA code, validated for Finnish power reactors and further elaborated for added applicability and flexibility. Parts of the code have been updated with models that represent the recent experience and experimental findings. Specific material properties, particularly those for VVER reactors, have been incorporated in the code. When the local burnup in a fuel rod exceeds 40 or 50 MWd/kgU, the models of burnup dependent phenomena like thermal conductivity, fission gas release, porosity and swelling, turn out to be more or less insufficient. More physically based models should replace the temporary correlations that have introduced to compensate this deficiency as soon as results of experimental and theoretical studies allow.

2. The Monte Carlo (MC) Method

In deterministic analyses, the effects of changing operation conditions are estimated by making computer runs with chosen fixed sets of initial conditions. For safety analyses it is conventional to define a worst-case or “conservative” selection of initial design and operating conditions. Sometimes this stacking of most unfavourable assumptions may be even unrealistic, and in any case, no information is gained on the relative importance of the parameters in the whole reactor scale. Recent advances in computer technology are rapidly increasing flexibility in applications, and probabilistic techniques, well developed in theory, become within practical reach.

In the probabilistic MC method, one takes known distributions of fabrication parameters and real power histories, and makes a large number of calculations with randomly selected combinations. Good statistics may require thousands of computer runs. The MC enables simulation of processes by generation of random variables provided that the distributions, Gaussian or other, are mathematically defined or can be readily simulated.

3. Quasi-Random Sequences

From literature, a procedure known as *Sobol’s quasi-random sequences* is used. Compared with standard random number generator of normal MC method this allows sampling with improvement of convergence rate of N^{-1} over $N^{-½}$, N being the number of computer runs.

The effect of quasi-random sampling is visualised in Fig. 1, which compares the random selection and Sobol’s selection.

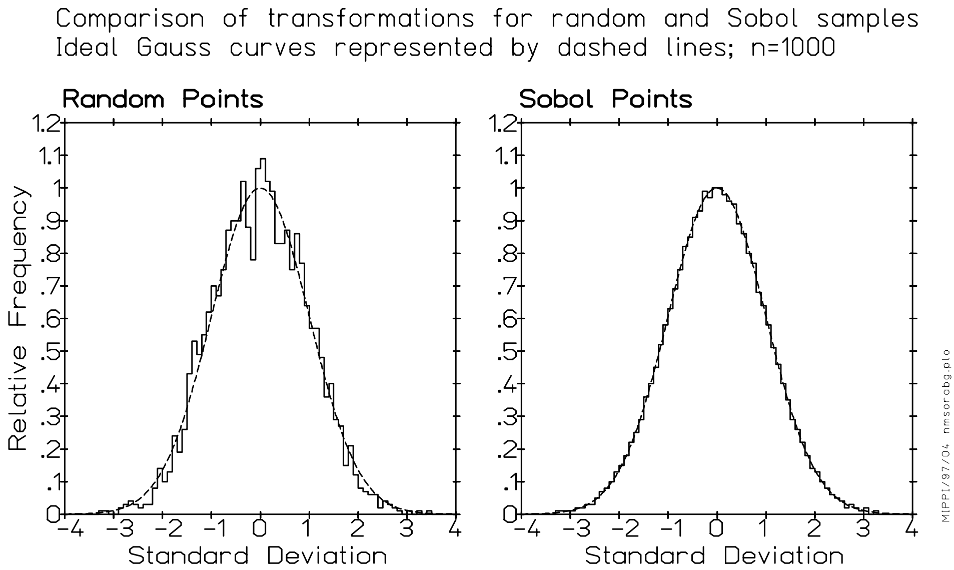


Fig. 1. Comparison of the random and quasi-random sampling methods.

4. VVER sample case input

In this example the statistical variation was implemented in eight dimensions. The variation was generated for four fabrication parameters, three model parameters and instantaneous power. The analysis was divided into two parts: 1) whole core assessment and 2) the worst case estimation. The numbers of computer runs were 3546 and 1000, respectively.

The VVER-440 rod **power histories** are obtained with the HEXBU-3D code. The core is loaded with 313 fuel assemblies. Each rod is subdivided into ten axial zones for which the power and coolant temperature are stored at 24 time steps during an operation cycle. Once we have the power histories for all rods, a pre-processor randomly selects the rods for the whole core ENIGMA analysis. Due to symmetry, a 30-degree sector sufficiently represents the core. Of each assembly that falls in the sector, 50 rods (25 from boundary bundles) are picked. At each time step, the rod power is multiplied by a normally distributed coefficient with a standard deviation of 5 % to describe the uncertainty in power determination.

Necessary distributions for **fabrication parameters** are produced with Sobol's generator again. The parameters vary independently. Varied become the following: **pellet radius, clad inner diameter**, **fuel grain size** and **fuel initial density.**

The fuel phenomena that become increasingly important with burnup are fission gas release (FGR) and changes in fuel thermal conductivity. In addition to the possible gas release from the high temperature pellet centre, there may be athermal release from the porous high burnup rim region. Three **key model parameters** were varied randomly within pre-defined limits.

The mechanisms of fission gas release (FGR) are known as far as low or medium burnups are concerned. However, the FGR processes have a strong temperature dependence and their dynamics remain a challenge. Therefore, the **diffusion coefficient** is permitted to vary, reflecting the uncertainties in fuel microstructure and others (Fig. 2).

High burnup fuel will liberate **fission gases** even **at low temperatures**. This is related to the UO_2 structure changes when the local burnup exceeds 60–65 MWd/kgU. At the moment, the phenomenon has been coded in ENIGMA only indirectly.

Fuel temperature (in-pile) and laser flash measurements (out-of-pile) have shown that **UO_2 thermal conductivity** steadily decreases with increasing burnup. A recent Halden experiments suggest a 40 % reduction (1000 K) at 65 MWd/kgU from the original value. The ENIGMA correlation, largely based on the evaluations of the Risø Fission Gas Projects' experiments, gives even more pronounced degradation in the conductivity. The degradation factor was randomly selected between these two extremes, which are plotted in Fig. 3.

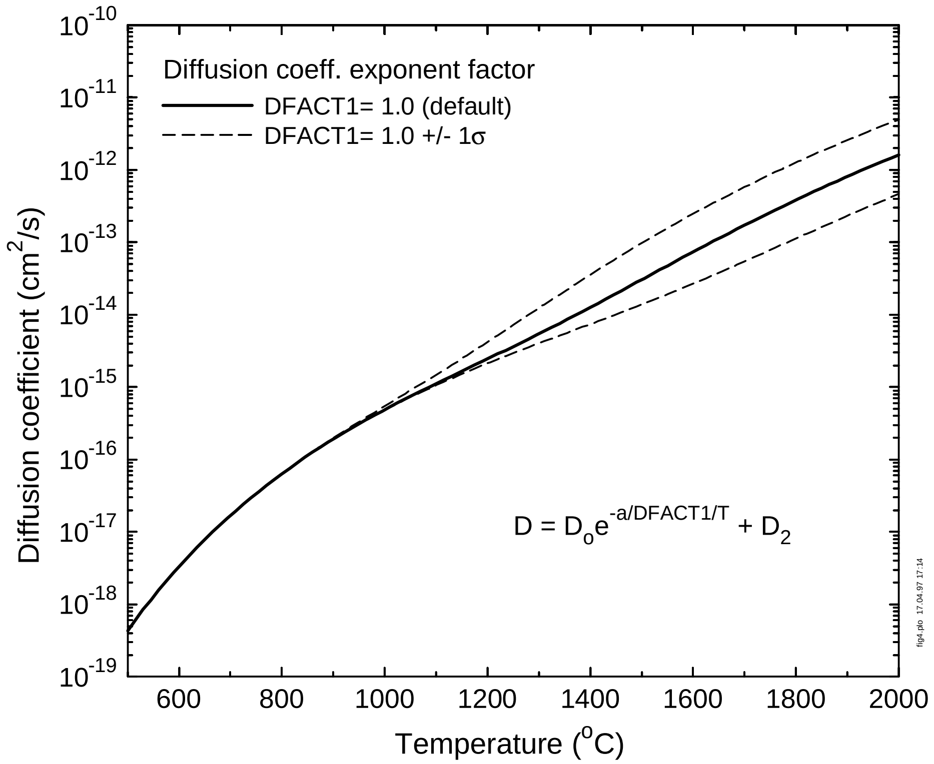


Fig. 2. Fission gas diffusion coefficient and its variation.

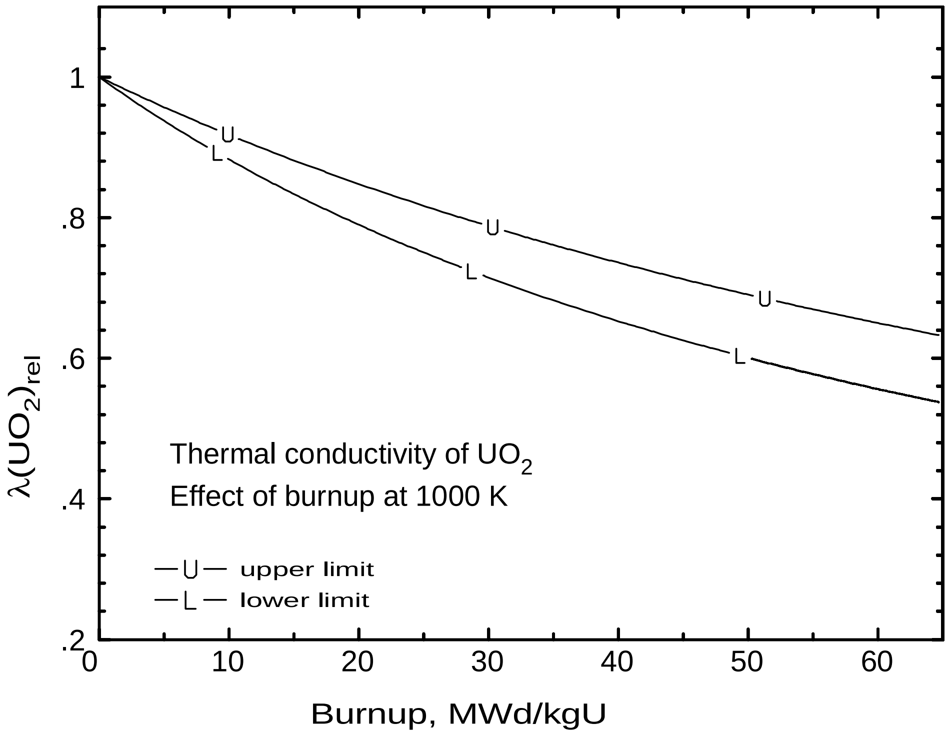


Fig. 3. Relative thermal conductivity of UO_2 ; upper and lower estimates.

5. Sample case results and discussion

The 1182 sampled power histories and the 3546 different calculated cases are expected to represent the state of the reactor during one hypothetical cycle with good accuracy. The highest fuel rod burnup does not exceed 50 MWd/kgU.

Relatively low **fuel temperatures** are predicted for all cases. The peak temperatures do not exceed 1200 ^{o}C. In the rods in their last cycle, temperatures are below 1000 ^{o}C. At the temperatures predicted by ENIGMA, the fission gas release is largely athermal. Comparison with the threshold curve derived in the Halden Project shows a margin of at least 100 ^{o}C to the thermal FGR.

The **athermal gas release** is important above about 40 MWd/kgU by gradually increasing the rod inner pressure. In hot conditions the pressure is typically below 2.5 MPa, but may in some cases be 3 MPa, which still leaves a large margin to the system pressure of 12 MPa, Fig 4.

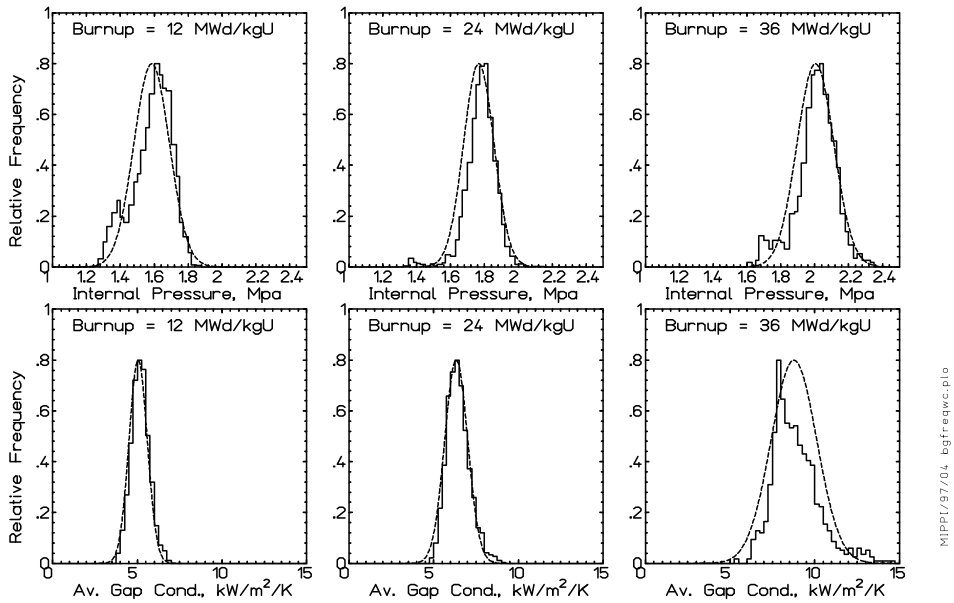
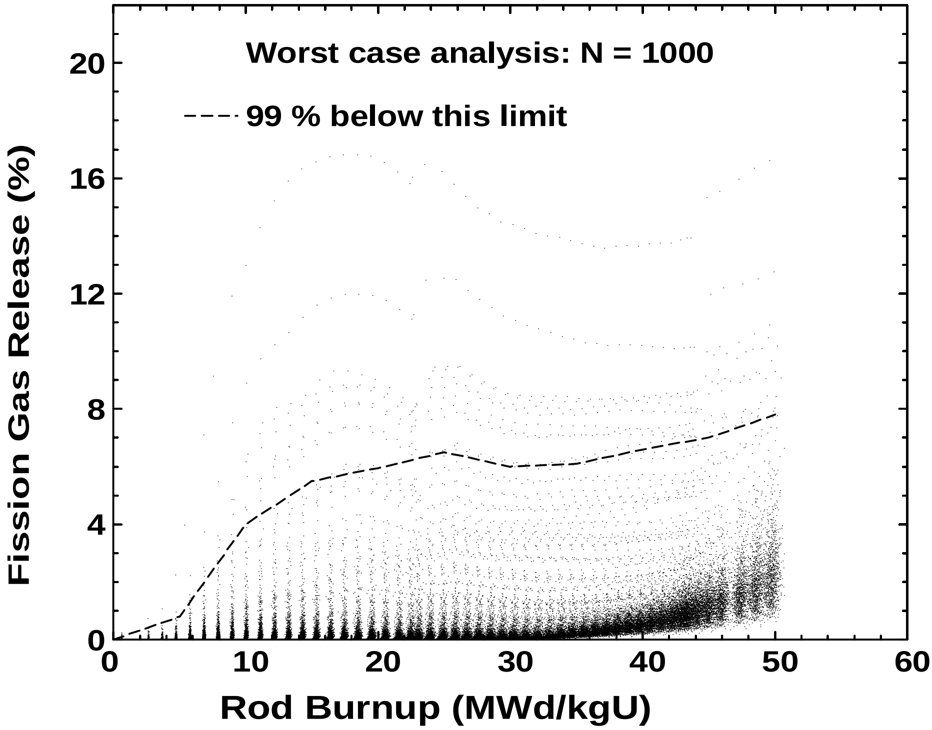


Fig. 4. Calculated rod internal pressure and gap conductance in the whole core.

Gap conductance depends on the gap width and gas composition. Generally the conductance seems to increase with burnup, due to the closing gap and lack of significant gas release. The scatter becomes gradually larger among second and third cycle rods. In the majority of cases the gap conductance still increases after second reactor-year.

A simple counting routine gives the frequency distributions for each variable within successive burnup steps. The technique can be used either to draw limiting curves (like in Fig. 5) or to calculate tolerance intervals by fitting a mathematical distribution function to the data.

Fig. 4 presents the relative frequency data for two output variables: Average gap conductance and rod internal pressure, plotted for rod average burnups of 12, 24 and 36 MWd/kgU.



Fi.g 5. Calculated variation of FGR.

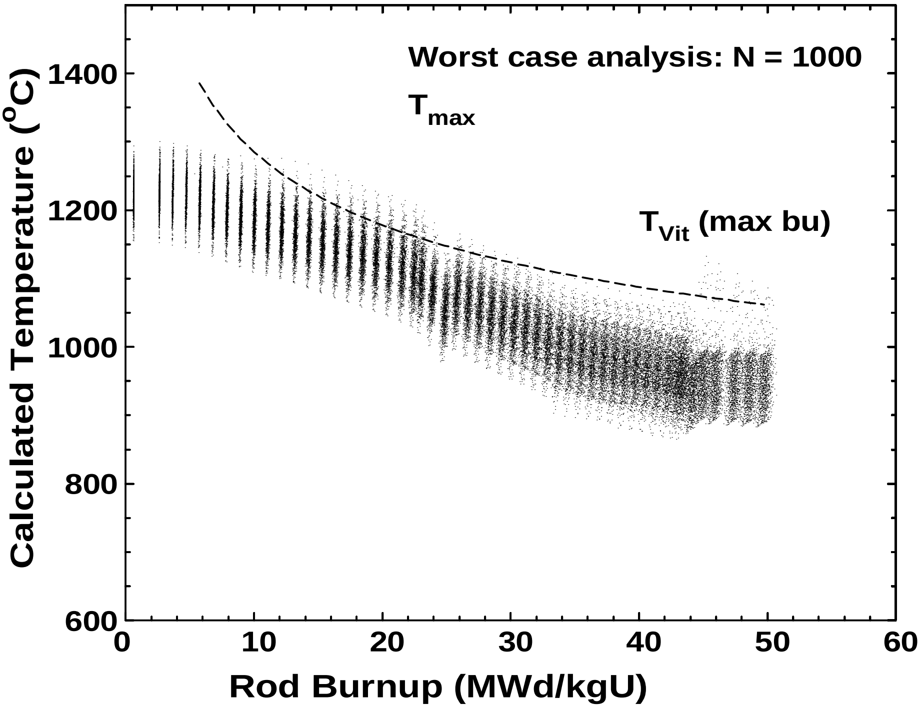


Fig. 6. Calculated variation of fuel centerline temperature.

6. Estimates for the Fuel Rod Parameters

Figs. 5 and 6 display the statistical results of the worst case analysis. The input parameters have their distributions as before, but in all runs the same power history, the highest allowed, is used. The axial position of the peak power zone moves as in one real rod. Within the 1000 cases studied, still only few indicate thermally activated gas release. The average end-of-life FGR is 2.5 %, whereas 5 % will be exceeded in 50 cases out of 1000, Fig 7. At the end-of-life, high value tails become visible unlike in the whole core distributions.

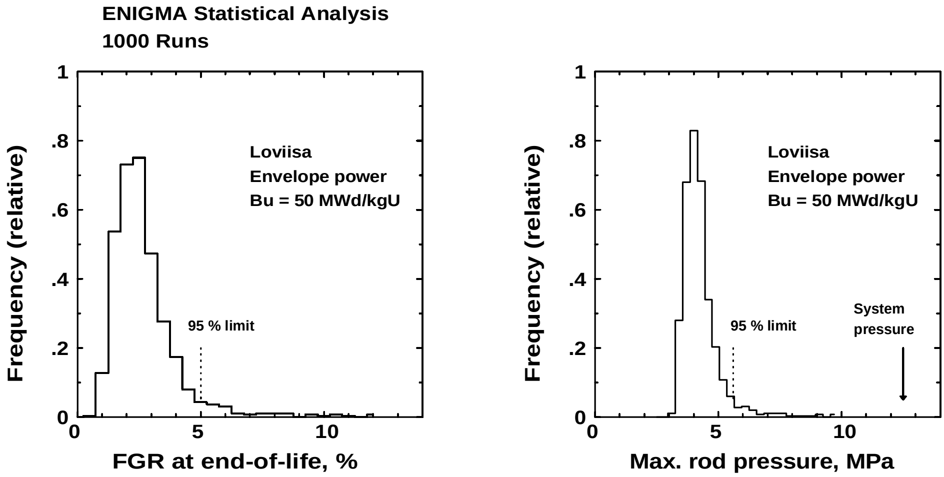


Fig. 7. Distributions of the calculated gas release and pressure.

7. Conclusions

The steady state fuel behaviour code ENIGMA was modified for flexible use in probabilistic analyses. Instead of the usual deterministic way with fixed sets of initial conditions, one can take known distributions of fabrication parameters and realistic power histories, and make a large number (of the order of 1000) calculations with randomly selected parameter combinations. This approach allows to find out the state of a whole reactor in terms of relevant fuel rod parameters, gap conductance, fission gas release and internal pressure in the first place. A procedure known as quasi-random sampling was coupled with the Monte Carlo method to improve the convergence rate, and thus permitting fewer computer runs. A sample case with VVER-440 reactor fuel indicated relatively low fuel temperatures and mainly athermal fission gas release at end-of-life. The rod internal pressure remains practically in all cases below 3 MPa, which leaves a large margin to the system pressure of 12 MPa. Due to clad creep-down and insignificant gas release, gap conductance typically increases with increasing burnup.

Development of the 3D BWR dynamics code TRAB-3D

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Abstract

TRAB-3D, the newest member of VTT's code system for LWR dynamics calculations, is a coupled neutronics-thermal hydraulics code for transient and accident analyses of BWR reactors. The code is largely based on the 1D code TRAB and the 3D hexagonal code HEXTRAN which have long been used in safety analyses of Finnish and foreign reactors. In TRAB-3D the two-group neutron diffusion equations are solved in three dimensions in a rectangular fuel assembly geometry by a new method which is similar to the nodal expansion method developed earlier at VTT for hexagonal geometry. The accuracy of the method is shown by comparison with 2D fine-mesh calculations and with 3D calculation for the Olkiluoto reactor with the POLCA-4 code. Capabilities of the code in dynamic analyses is validated with the OECD/NEA LWR benchmark problems and with transient calculations for the Olkiluoto reactor. Further development of the code includes a pin power reconstruction method which makes use of precomputed power distributions within fuel assemblies.

1. Introduction

Development of the three-dimensional (3D) reactor dynamics code TRAB-3D (Kaloinen & Kyrki-Rajamäki 1997, Kaloinen et al.. 1998) complements VTT's calculation system for light water reactors. TRAB-3D is largely based on the 3D dynamics code HEXTRAN (Kyrki-Rajamäki 1995) for VVER reactors and on the axially 1D code TRAB (Rajamäki 1980). The neutron kinetics and thermal hydraulics models of the reactor core are equal to those of HEXTRAN and the calculation models for the BWR primary circuit are adopted from TRAB. For PWR calculations the circuit model of the SMABRE code (Miettinen 1985) can also be included in TRAB-3D.

The two-group neutron diffusion equations are solved in a rectangular geometry by a nodal expansion method which employs similar basic principles as the hexagonal method in HEXTRAN developed originally for the core design codes HEXBU and HEXBU-3D (Siltanen et al. 1974, Kaloinen et al. 1980). Comparison with fine-mesh finite-difference calculations and with calculation results of the fuel management code POLCA-4 for an Olkiluoto core have shown that the accuracy of the new method is more than sufficient for dynamic analyses. Although there are several advanced features in the solution method, e.g. the use of flux discontinuity factors, the computing times are reasonably short.

Development of the neutronic models has continued with the design of a pin power reconstruction method (Mattila 1998). The reconstruction makes use of pre-computed pin power distributions which are modulated by the intranodal distributions from the nodal solution to produce the actual pin powers. Test calculations for a checkerboard loading of four different fuel assemblies have given a good agreement with accurate calculations with the cell burnup code CASMO-4.

TRAB-3D has successfully been tested against the OECD/NEA LWR core transient benchmarks. For a PWR reactor these problems include several cases of control rod ejection accidents and a transient of slow withdrawal of control rods. The BWR benchmark consists of inlet cold water and core pressurization transients, but no results have been published for the latter case. To demonstrate the performance of the code when the whole BWR primary circuit is modelled, a pump trip transient has been calculated and the results compared to a one-dimensional calculation with TRAB.

2. Solution of neutron diffusion equations

The nodal method of TRAB-3D to solve the two-group neutron diffusion equations in a rectangular fuel assembly geometry is closely related to the hexagonal method of HEXTRAN. Main similarities of the methods are construction of group fluxes from two characteristic solutions or spatial modes and approximation of them with polynomials or exponential functions within nodes. Also, the description of homogenized cross sections are similar in the codes.

Starting point of the method is decoupling of the two-group diffusion equations and construction of group fluxes from the characteristic solutions. The fast and thermal flux are expressed as linear combinations of the two solutions

$$\Phi_1(r) \;=\; f_I(r) + R_{II} f_{II}(r) \qquad (1)$$

$$\Phi_2(r) \;=\; R_I f_I(r) + f_{II}(r). \qquad (2)$$

The spatial functions $f_I(r)$ and $f_{II}(r)$ are called the fundamental or asymptotic mode and the transient mode of the solution. Coefficients R_I and R_{II} depend on the nodal cross sections and on the effective multiplication factor. The fundamental mode is a smoothly varying function within a node, but the transient mode has a large and negative buckling in light water reactors and is (potentially) nonzero only near material discontinuities.

Behavior of the flux modes suggests different approximations for them within a node. The choice in TRAB-3D is polynomials for the fundamental mode and exponential functions for the transient mode. Furthermore, the fundamental mode is assumed separable in radial and axial directions within each node

$$f_I(r) \;=\; Af_{xy}(x,y)f_z(z). \qquad (3)$$

f_{xy} and f_z are the spatial shapes of the radial and axial components and A is an amplitude factor. The polynomial approximations for the shape functions are of fourth order in the radial coordinates x and y and of fifth order in the axial coordinate z. Altogether, the fundamental mode approximation contains ten degrees of freedom. This is less than the number of terms in the polynomials since low-order residuals of the decoupled diffusion equation are set to zero which gives additional relations between the terms.

The transient mode is approximated separately at each face of the node by an exponential function. Such an approximation can be considered acceptable, because the relaxation length of the mode is small compared to the typical value of 15-20 cm for the nodal dimensions. There is an equal number of degrees of freedom for both modes giving a total number of 20 per node in addition to the amplitude factor.

Relations to determine the unknown parameters in the approximations are based on average quantities at nodal interfaces. In a radial plane the continuity condition of flux and net current and of their first moments is imposed, but in axial direction only the flux and current are required to be continuous. In TRAB-3D the homogenized parameters can include flux discontinuity factors for the transverse faces of the nodes and these factors are embedded into the continuity relations.

The most important part of the nodal solution is the calculation of coupling coefficients which relate the interface current of neutrons to the node-averaged values of the fundamental mode. For adjacent nodes A and B the surface-averaged current of group g neutrons is given by

$$\bar{j}_g^{AB} \quad = \quad K_g^A \bar{f}_I^A - K_g^B \bar{f}_I^B . \tag{4}$$

K_g is the coupling coefficient and the bar denotes averaging over nodal surface or volume. The explicit shape of the flux within a node is not calculated since the coupling coefficients contain all information on it needed in solution of the global node-averaged distribution of flux.

The nodal equations are solved by an efficient two-level iteration technique where one unknown per node is improved in inner iterations. These iterations solve seven point equations which are constructed for the node-averaged fundamental mode by integration of the diffusion equations over nodal volumes and by use of coupling relations (4). During outer iterations the node flux shapes are improved or the coupling coefficients are recalculated. Also, the thermal hydraulic calculations are carried out and the nodal cross sections and values of kinetic variables are updated in outer iterations.

3. Pin power reconstruction

The pin power reconstruction scheme of TRAB-3D is based on modulating the local heterogeneous power distribution of a node with the global "homogeneous" power distribution interpolated from the solution of the nodal equations. In principle the modulation could be done for the fast and thermal fluxes separately

$$F_g(x,y) \quad = \quad F_g(x,y)_{hom} F_g(x,y)_{form} \tag{5}$$

and the global heterogeneous power would then be calculated from the reconstructed flux. This would, however, require knowledge of the heterogeneous cross sections within the node and such data is not easily available in a nodal code. Therefore the homogeneous fluxes are converted into the "homogeneous power" and the modulation is then done with the equation

$$P(x,y) = P(x,y)_{hom} P(x,y)_{form} \tag{6}$$

Of course, an error is made in assuming a common multiplicator for the fast and thermal fluxes in $P(x,y)_{hom}$, but this assumption is presumed justified since the power associated with the fast flux is comparatively small and the fast flux varies smoothly within a node.

The local heterogeneous power distribution is calculated with the CASMO-4 cell burnup code and tabulated by void, void history and burnup for each fuel assembly. The actual distribution corresponding to the physical state of the node is found by interpolation during the pin power reconstruction. The effect of a cruciform control rod (CR) on the heterogeneous power distribution is taken into account in three stages. Firstly, an assembly with a CR in its corner is treated as a separate fuel type. Secondly, the effect of a CR on the bundles next to the one by which it is situated is calculated with pin-wise correction factors that depend on burnup, void and void history. The history effect is treated as a third correction.

The history effect of a control rod on the pin powers depends on the time of insertion (increased plutonium production and decreased burnup in fuel pins close to the CR) and withdrawal (history effect diminishes). These effects are described by an eight-parameter function for the insertion and withdrawal time of the CR. The parameters are precomputed and tabulated according to void and void history.

The accuracy of the reconstruction method was tested in an infinite core of 2x2 checkerboard loading with Atrium fuel assemblies of four different burnups. The reference solution to the pin power distribution was calculated with CASMO-4 and a comparison was made between homogeneous power distributions created from the nodal solution of TRAB-3D and from the reference solution. Figure 1 shows the reconstructed and reference distributions for one of the fuel assemblies. Deviations of the reconstructed pin powers were found to be within 1 % of the CASMO-4 result.

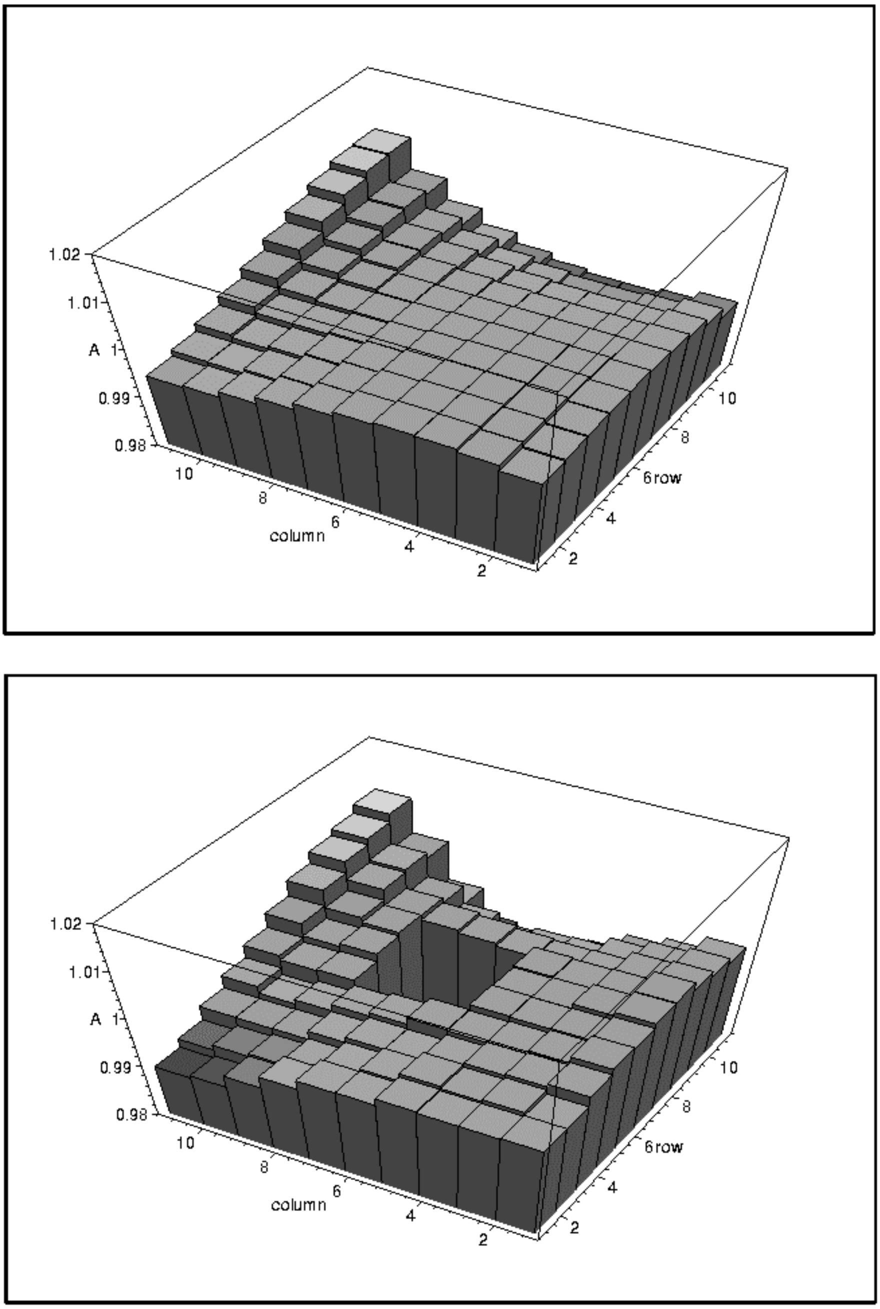

Figure 1. Reconstructed (above) and reference (below) distributions of pin power in a BWR fuel assembly.

4. Calculation of OECD/NEA benchmark

The first dynamic calculations with TRAB-3D were performed for the NEA 3-D LWR core transient benchmark which includes both PWR and BWR problems (Finnemann & Galati 1992, Finnemann et al. 1993). Calculation of the various PWR cases is described in technical reports and in the Interim Report of the research programme (Vanttola & Puska 1997). In general, the TRAB-3D results were in a good agreement with the reference solution calculated with the PANTHER code. As an example of the results, Figure 2 illustrates the highly asymmetric power distribution at the time of power maximum calculated for one case of the control rod ejection transients. Such a transient, where a single peripheral rod is ejected, imposes strict requirements on the calculation models even if there is no coolant boiling in the core.

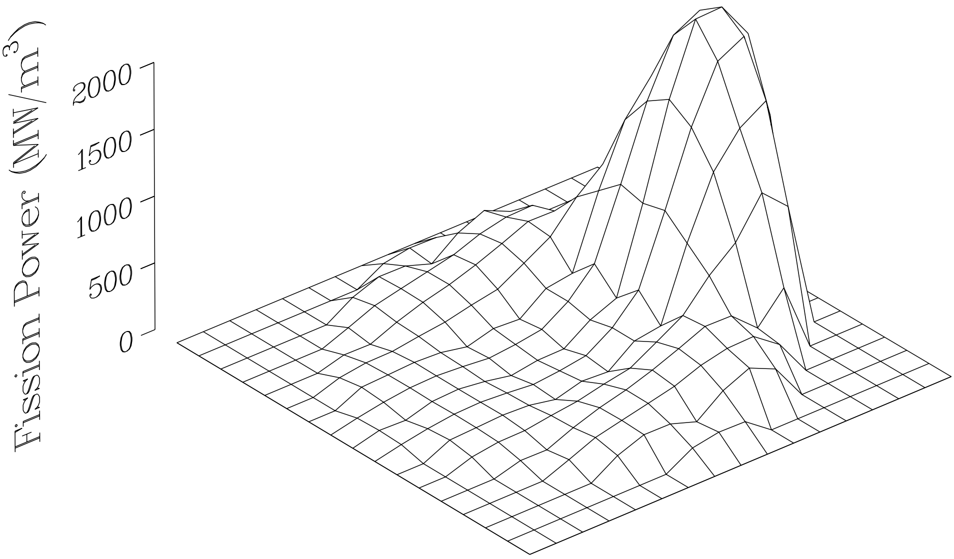

Figure 2. Core power distribution at time of maximum power for a control rod ejection transient.

The two BWR problems included in the benchmark are a cold water injection transient and a core pressurization transient, but for the present, results of only the first case are reported in open literature. The original definition specified a very fast exponential increase of pressure with a 0.04 s time constant. Later a more realistic problem was defined by increasing the constant to 0.4 s and this slower transient was calculated with TRAB-3D. Some results are given in the following.

The calculated value of 0.9868 for the effective multiplication factor in the initial steady state agrees well with the mean of 0.9859 given in the report of the benchmark results. Also, the distributions of coolant flow and outlet density are close to the distributions around which most of the reported results are clustered. However, TRAB-3D tends to give for the axial power a less downward peaked distribution than the other calculations.

The disturbance in first case is an exponential decrease of inlet water temperature by 9 K with a 2.5 s time constant and in second case a pressure increase of 13 bar with a 0.4 s time constant. In TRAB-3D the shortest time steps were 10 ms around the power peak. Figures 3 and 4 show the time behavior of the total reactor power and core-averaged fuel temperature for the two BWR cases.

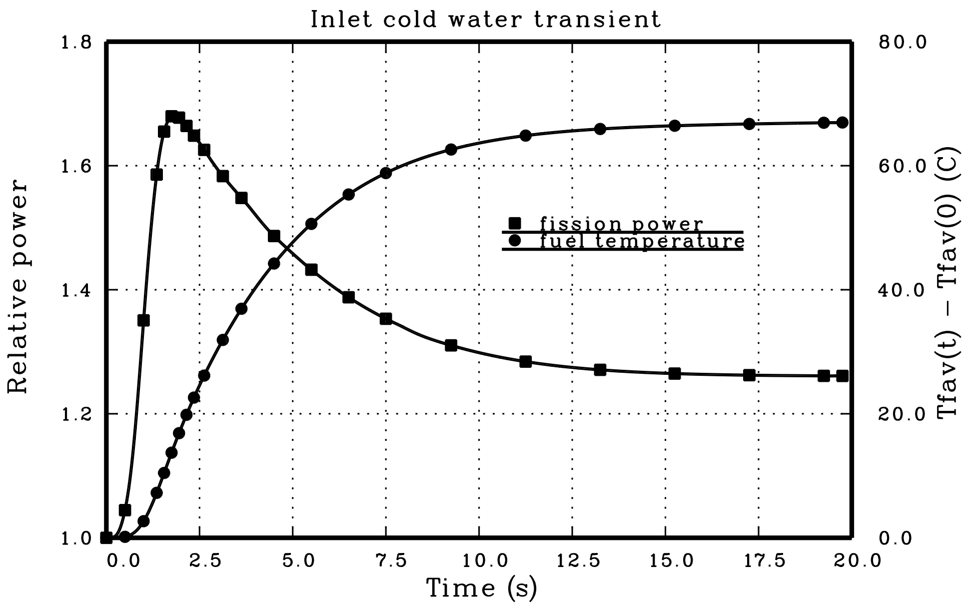

Figure 3. First BWR benchmark, a cold water injection transient. Fission power and fuel temperature increase from steady state by TRAB-3D.

The reported results of seven codes for the first case give values of power maximum which vary from 1.3 to 2.1 (relative to initial power). Thus the calculated maximum of 1.67 is roughly equal to the average of these results. A similar comparison is found for the time behavior of fuel temperature; again the TRAB-3D curve is close to the average of other curves.

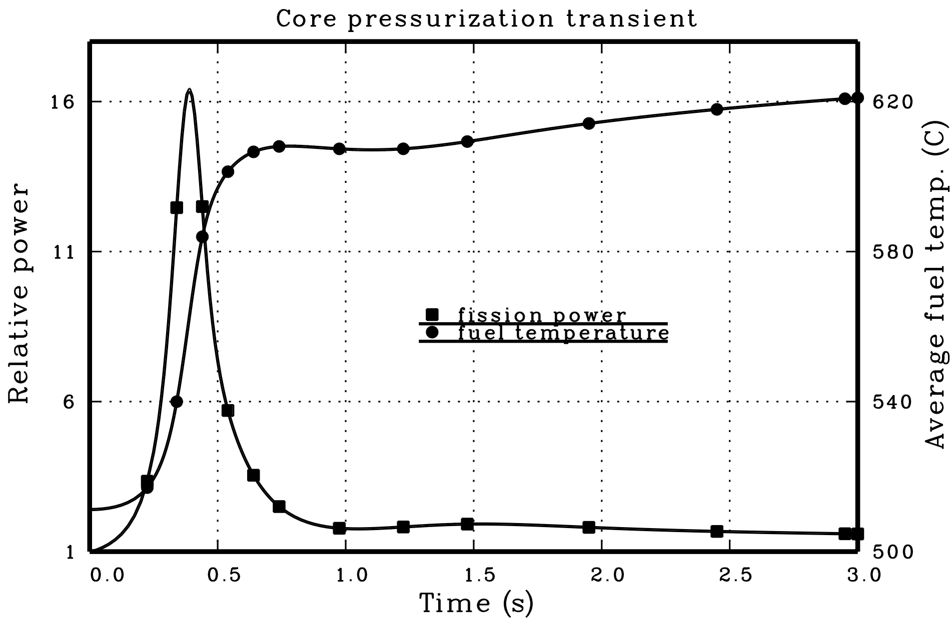

Figure 4. Second BWR benchmark, a core pressurization transient. Fission power and average fuel temperature by TRAB-3D.

Results for the second case can be compared to one-dimensional TRAB calculations made earlier at VTT (Räty et al. 1993) and to recent calculations with the three-dimensional code DYN3D/R (Rohde 1998). TRAB-3D gives a value of 16.4 for the maximum of relative power which is somewhat lower than the values of 21.7 by TRAB or 23.3 by DYN3D (1 ms time step). On the other hand TRAB-3D predicts a slightly larger increase in fuel temperature than DYN3D, but a lower increase than TRAB. This calculation demonstrates the applicability of TRAB-3D for analyses of overpressurization events.

5. Calculation of the Olkiluoto reactor

The test case for the Olkiluoto reactor consists of an equilibrium loading designed with the POLCA-4 code at the TVO power company. Two types of Siemens Atrium assemblies are loaded into the core and the assembly burnup varies between 0 - 37 GWd/tU in the beginning of cycle.

A steady state calculation at the full reactor power of 2500 MW gave a very good agreement with the POLCA-4 results if unity discontinuity factors are used in TRAB-3D. According to the calculations the arithmetic average of deviations in assembly powers is 0.8 % and the maximum deviation is as low as 2.2 %. At the location of maximum assembly power the deviation is less than half a per cent. Around inserted control rods the assembly powers of TRAB-3D tend to be systematically higher by 1-2 % than the values of POLCA-4 which could be partly explained by slightly different modelling of control absorbers in the codes.

The results differ much more from each other when the TRAB-3D calculation is made with assembly discontinuity factors, because POLCA-4 does not model the flux discontinuity at nodal interfaces in solution of the neutron diffusion equations. Test calculations showed that discontinuity factors have a fairly large effect on the calculated power distribution. In general, the power decreases in assemblies of inserted control rods where the discontinuity factors deviate most from unity. Because the decrease is 6-7 % for assemblies of fully inserted rods, the deviations from the POLCA-4 result are larger than for the TRAB-3D calculation with unity discontinuity factors. However, the real accuracy of the different results is uncertain in this case when no comparison has been made against measurements.

To make realistic calculations, the BWR circuit model of TRAB was connected to TRAB-3D. A schematic illustration of the model is shown in Figure 5. The circuit models are one-dimensional, but include a number of parallel components.

A pump trip transient initiated by loss of auxiliary power was calculated as a first dynamic test for the entire TRAB-3D code with both three-dimensional core and BWR circuit model. In the calculated case, the speed of the recirculation pumps starts to decrease due to loss of auxiliary power. One group of the pumps trips by losing the electric torque, and another group, the pumps with flywheel, coasts down following a given ramp for the pump speed. Simultaneous reduction of feedwater flow leads to turbine trip after a delay, and reactor steam flow is released through dump valves. Mismatch in the operation of turbine and dump valves results in a rapid increase in the reactor pressure which is restored to the initial condition by the pressure control system.

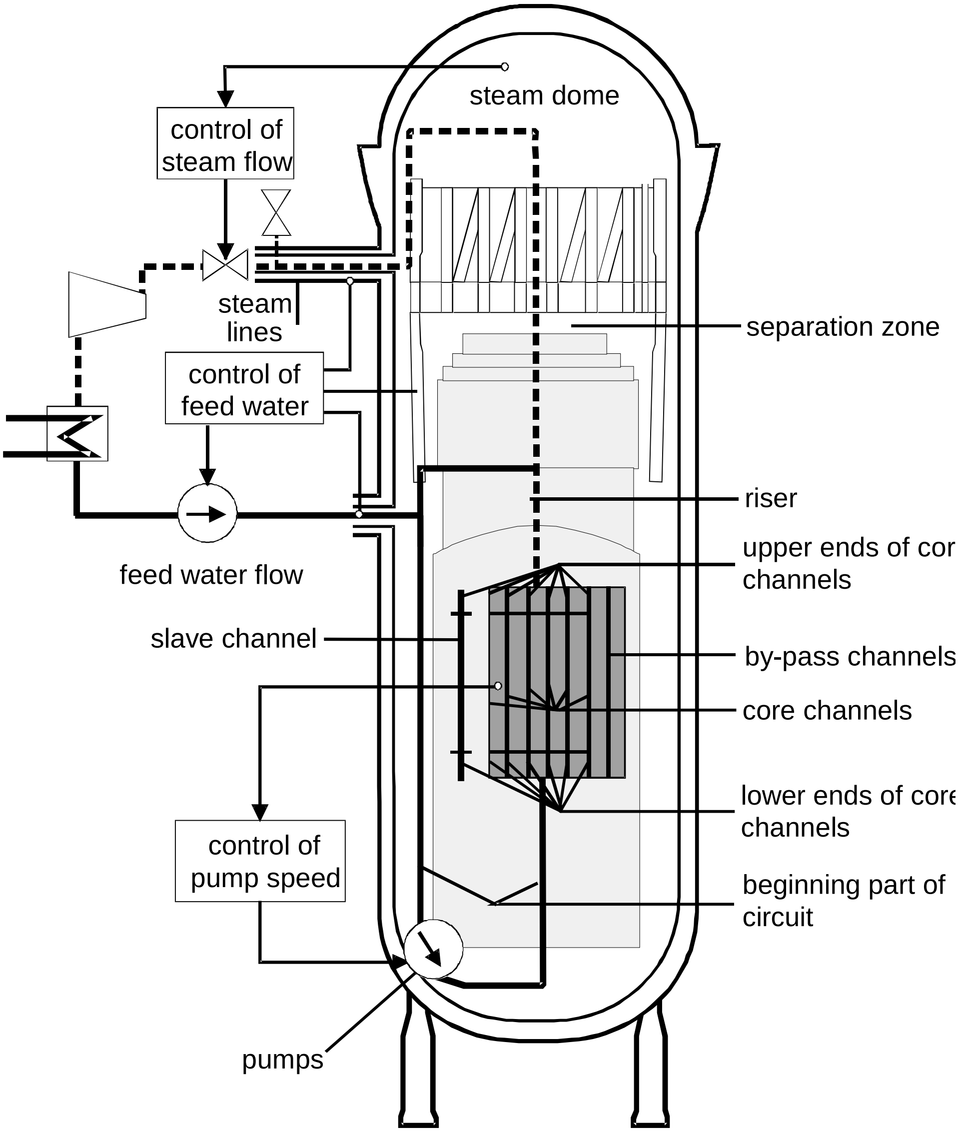

Figure 5. Components of BWR circuit models in TRAB-3D.

For comparison, a corresponding calculation was made with the TRAB code where the neutronics model is axially one-dimensional. The circuit modelling in both calculations, the geometry in the fuel and flow channels, and the thermal hydraulic correlations were identical. In the one-dimensional calculation the void and fuel temperature coefficients as well as the axial power distribution were adjusted to match the values of the three-dimensional calculation. Figure 6 shows the time behavior of the fission power and mass flow in the downcomer predicted by the two codes. Results of the three- and one-dimensional calculations agree very well, as they should in such a transient with no prominent three-dimensional effects in the core. The test calculation demonstrates the capability of TRAB-3D in analyzing the coupled behavior of a BWR core and cooling circuit. Although in this particular transient, a three-dimensional core model may seem superfluous, it would allow the accurate modelling of mixed cores consisting of different types of fuel assemblies. Subsequent calculations for e.g. overpressurization transients will demonstrate the full worth of the three-dimensional core modelling.

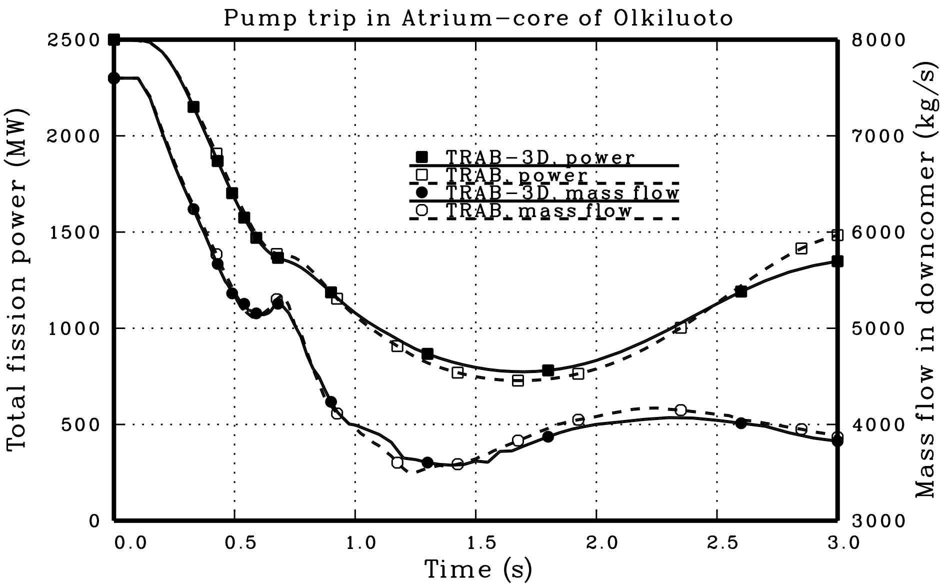

Figure 6. Pump transient from full power for an Atrium core of Olkiluoto BWR. Comparison between three-dimensional and one-dimensional results.

6. Conclusion

A new code TRAB-3D was developed at VTT for three-dimensional dynamic analyses of light water reactors with rectangular core geometry. Thermal hydraulic models for the reactor core and cooling circuits in the code are adopted from existing codes which have already been validated and applied for several years at VTT. The two-group neutron diffusion equations are solved in three dimensions with a sophisticated method which has an accuracy comparable to the accuracy of advanced nodal methods in static core design codes. Still the code is fast enough for analyses of longer transients with 3D neutronics.

The most recent improvement of TRAB-3D is the development of a pin power reconstruction method which will be an integral part of the code. For the present the method has been successfully tested against detailed transport calculations with the assembly code CASMO-4.

All problems of the NEA LWR core transient benchmark were successfully calculated with TRAB-3D. Generally, the results compared favorably with other calculations for which there are published results. Since the benchmark consists of PWR and BWR transients accurate calculations require correct models for both core neutronics and thermal hydraulics. Results of a steady state calculation for a planned equilibrium core of the Olkiluoto reactor gave an excellent agreement with results of the fuel management code POLCA-4 when unity discontinuity factors are applied in TRAB-3D.

The dynamic capability of the entire TRAB-3D model was demonstrated by analyzing a pump transient in which the whole primary circuit of a BWR reactor is included in the calculation. To verify the results, the same transient was calculated with the one-dimensional code TRAB which has been validated earlier. Good agreement of the results confirms the correct functioning of TRAB-3D in analyzing the transient behavior of a BWR core coupled with cooling circuit.

The next step in verification and validation of the code is the analysis of real BWR transients occurred in the Olkiluoto power plant. At the same time NEA’s Main Steam Line Break benchmark will be calculated with a code system which consists of the TRAB-3D core model coupled with the PWR circuit model

SMABRE. As a future development, VTT's new and accurate thermal hydraulics solver PLIM will be applied also in TRAB-3D (Rajamäki et al. 1998).

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Application of an accurate thermal hydraulics solver in VTT's reactor dynamics codes

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Abstract

VTT's reactor dynamics codes are developed further and new more detailed models are created for tasks related to increased safety requirements. For thermal hydraulics calculations an accurate general flow model based on a new solution method PLIM has been developed. It has been applied in VTT's one-dimensional TRAB and three-dimensional HEXTRAN codes. Results of a demanding international boron dilution benchmark defined by VTT are given and compared against results of other codes with original or improved boron tracking. The new PLIM method not only allows the accurate modelling of a propagating boron dilution front, but also the tracking of a temperature front, which is missed by the special boron tracking models.

1. Introduction

VTT's calculation system for reactor dynamics consists of the axially one-dimensional TRAB code (Rajamäki 1980, Räty et al. 1991) and the three-dimensional codes TRAB-3D (Kaloinen & Kyrki-Rajamäki 1997) for square lattice geometries, and HEXTRAN (Kyrki-Rajamäki 1995) for hexagonal geometries. All the codes include full circuit models. HEXTRAN has been thoroughly validated and extensively applied to safety analyses of the VVER-440 type Finnish NPP Loviisa, the Hungarian VVER-440 type NPP Paks, as well as the new Russian concept VVER-91. TRAB-3D is a new code presently being validated (Kaloinen et al. 1998a, 1998b); its one-dimensional predecessor TRAB has for long been utilized in transient analyses of the Finnish Olkiluoto BWRs.

In LWR accident analyses many complicated phenomena arise which cannot be correctly solved with the hydraulic solution methods applied so far in the world. Especially reliable calculation of propagating boron fronts is very difficult with standard numerical algorithms because numerical diffusion tends to smoothen the front. In this way, the reactivity effect of the boron dilution can be significantly lowered and conservatism of the analyses cannot be guaranteed. The new hydraulics solution method PLIM, Piecewise Linear Interpolation Method (Rajamäki & Saarinen 1994a, Rajamäki 1996, 1997), is capable of avoiding the excessive errors, the numerical diffusion and also the numerical dispersion.

Application of the PLIM Method in VTT's 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 (Separation of Flow According to Velocity, eg Rajamäki & Saarinen 1994b, Narumo 1997) 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 computational fluid dynamics code CFDPLIM, which solves a general multi-variable system of flow equations.

2. The hydraulics solver CFDPLIM and its application to reactor dynamics

PLIM, Piecewise Linear Interpolation Method, is a new highly accurate shape-preserving method for solving systems of one-dimensional hyperbolic partial differential equations. It uses a concept of characteristics. PLIM is applicable and accurate always when conventional methods are accurate and is able to treat advancing piecewise linear distributions accurately in any 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.

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. PLIM method has been successfully tested in several demanding flow problems, eg stratified two phase flow, gas dynamics and various convection diffusion problems (Saarinen 1994).

The solver CFDPLIM solves the system of N flow equations in an arbitrary hydraulic network, which is composed freely of nodes and one-dimensional flow paths. In addition to the normal input, the user has to define the terms of the equations in his/her own subroutines. The nodes with finite volumes, the boundary conditions of the flow paths, and geometric or moving discontinuities are also defined with functions written by the user.

The objectives of the development work of CFDPLIM, apart from those satisfied already by the use of PLIM, are

- It solves the unknown quantities when the terms of the equations are correctly defined as functions.
- The geometry of the network is not restricted in any manner. However, the components of the network can be arranged in particular orders to produce better convergence.
- The order N of the equations is arbitrary and it can be different in different parts of the network. If N is large and all of the variables are not strongly coupled, the system of the equations can be divided into sub-systems.
- Parallel computation can be utilized in evaluation of the function values. Hence rather complicated forms of the terms of the equations may be used.
- As few algorithms as possible are employed in CFDPLIM. Hence diffusion-like terms and discontinuities are assumed to be a part of the normal computation required for the mesh cells. Also this property offers an opportunity to apply parallel computation efficiently.

The application of PLIM in VTT’s reactor dynamics codes is carried out concurrently for the codes in production use, three-dimensional HEXTRAN and one-dimensional TRAB, which have identical thermal hydraulics models in the core channels. Calculation of heat transfer, neutronics and hydraulics is reorganized in order to solve the whole hydraulic circuit separately. The work consists of both representing the physical description in a form applicable to PLIM and of the necessary transfer of data between the new and original parts of the code and solver in a relevant form. The loose coupling between the original dynamics model and CFDPLIM allowing a smooth introduction of new code or solver versions is illustrated in Figure 1.

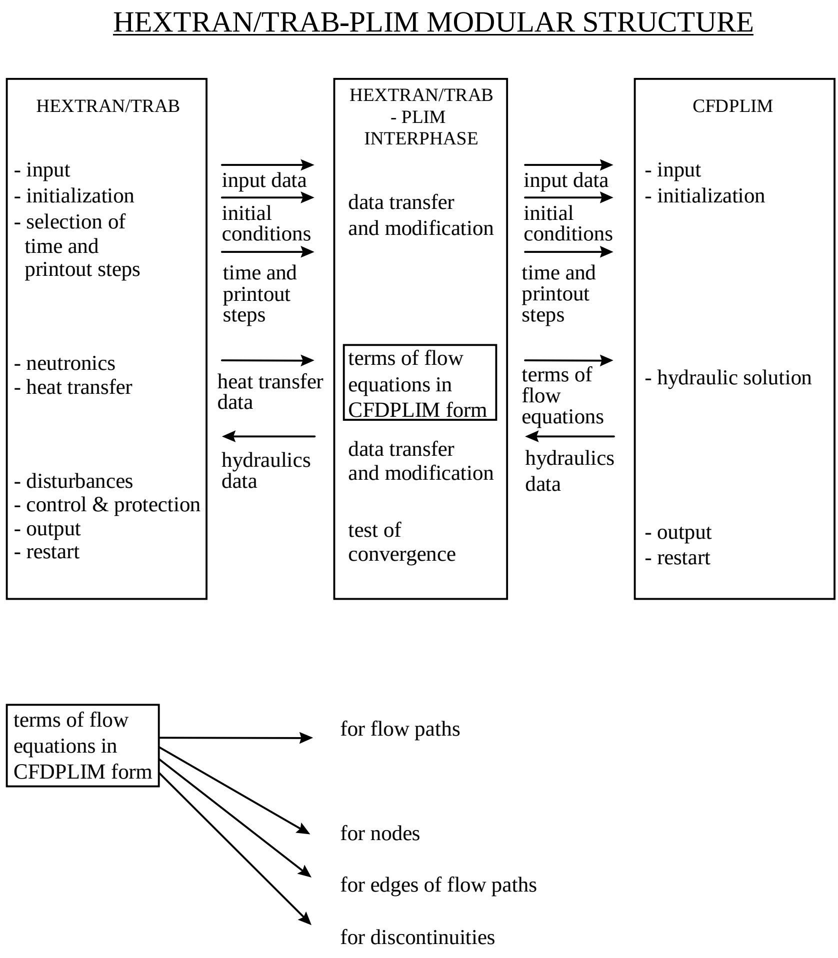


Figure 1. Modular structure of HEXTRAN-PLIM or TRAB-PLIM.

Early testing of CFDPLIM in the dynamics codes was carried out 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 (Rajamäki & Kyrki-Rajamäki 1997). 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 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. Most recently a boron dilution benchmark including boiling in the core channels has been calculated with the three-dimensional HEXTRAN-PLIM model. The results will be discussed in the next two chapters.

3. Three-dimensional hexagonal boron dilution benchmark in a VVER core

Local boron dilution has in recent years been discovered to be a possible cause of Reactivity Initiated Accidents. However, its accurate calculation with conventional numerical thermal hydraulics methods is not straightforward especially in low flow velocity conditions. Therefore a benchmark problem has been created in which the local boron dilution disturbance arrives into the core with the velocity given by one running reactor main coolant pump which is a probable situation in hot standby conditions after the restart of the first pump. The natural circulation conditions were considered too difficult for a benchmark calculation.

This benchmark problem (Kyrki-Rajamäki 1996) was the fourth one in the series of hexagonal three-dimensional dynamic benchmark problems which have been defined in the AER cooperation on VVER reactor physics and reactor safety. The first three problems were control rod ejection cases (Kyrki-Rajamäki et al. 1996). The local boron dilution accident case was calculated in a realistic VVER-440 core with fuel of three different enrichments. It concentrated mostly on the interaction of neutron kinetics and thermal hydraulics modelling of the core. The participants used their own nuclear data but the thermal hydraulic conditions at the inlet of the reactor core were given thus eliminating the need of circuit calculations.

The initiating event of the benchmark was a boron dilution and a coolant temperature disturbance in hot subcritical state of the reactor when all control rods were fully inserted. The transient occured in BOC conditions with a high content of soluble boron in the coolant. In the problem a reactivity excursion was calculated resulting in a prompt fission power peak. The flow condition was such that one reactor main coolant pump out of six was running. Thus the effect of the thermal hydraulics model on the transient was maximized yet avoiding the numerical difficulties in modelling of natural circulation flow. The flow conditions were approximated with 60^{o} symmetry in the core.

Solutions were received from VTT Energy from Finland with the HEXTRAN code, KFKI Atomic Energy Research Institute from Hungary with KIKO3D, Kurchatov Institute from Russian Federation with BIPR8/ATHLET, Research Centre Rossendorf from Germany with DYN3D and Nuclear Research Institute Øe• from the Czech Republic with DYN3D. The results have been summarized by Kyrki-Rajamäki (Kyrki-Rajamäki 1998a, 1998b).

In order to be able to properly calculate the benchmark new special boron front tracking methods were developed to the codes DYN3D, KIKO3D and BIPR8/ ATHLET to reduce the effects of numerical diffusion on the boron dilution front. The new general hydraulics solution method CFDPLIM was applied in HEXTRAN parallelly to the present production version 2.7 of the code.

As the nuclear data was not given in this benchmark problem, there was a need to fix some key parameters of neutronics so that there would not be too large differences between the initial states and between the reactivity levels caused by the disturbances. The need for tuning was in most cases not very large but there were some prominent exceptions.

The disturbance was the decreasing of boron concentration with 1014 ppm and coolant temperature with 30 °C at the core inlet at time 1 s with a ramp of 1 s. The cold diluted slug had a volume of 8.5 m^3 and after it had entered the core there was again a ramp of 1 s back to the original values. The static reactivity levels before and after this disturbance were tuned to be the same by every participant. Originally the static overcriticalities due to the disturbances reached from -2700 pcm to +2270 pcm; thus no meaningful comparison of the results would have been possible without the tuning.

The static overcriticality value after the disturbance is the most important parameter defining the results of the transient. Its value is defined to be 2270 pcm which belongs to the super-prompt critical region. By attaching its value in the benchmark it was possible to concentrate on the differences in the results due to the calculation of boron front travelling in the core. However, the exercise was still far from straightforward. The reactivity level of the core during the transient is very sensitive to the effective boron concentration and also to its distribution in the core. The whole reactivity value of the disturbance is 12550 pcm but only 18 % of it belongs to the supercritical region. An error of eg 3 % in the integral effect can cause a deviation of 17 % in the overcriticality with considerable effects on resulting prompt power peak.

The results of the benchmark are compared in Figures 2 - 6. The HEXTRAN results here are calculated with the production version yet without the PLIM solution. In radial power distributions the agreement between the results is better in the axially middle part of the core than at the edges; there seem to be larger differences in the modelling and data of reflectors than in the fuel area itself. In Figure 2 is an intersection near the core center in the steady state. As can be seen also the two solutions with the same code DYN3D differ slightly from each other because different nuclear data libraries were used.

In Figure 3 the average boron concentrations in the active core can be seen. Due to the numerical diffusion the theoretical maximum dilution of -1014 ppm from the initial value may not be achieved at all. If the smoothening effect is large enough, the whole transient can vanish because clear overcriticality is not reached. The fission power peak of the first preliminary solution calculated with DYN3D without the new particle-in-cell boron tracking method was smaller by a factor of 11 and occurred much later than the peak of the final solution. With BIPR8/ATHLETE the first and last fission power peak results differed with a factor of $3.4 \; 10^{9}$ showing that this type of accident could not at all be calculated with the standard model using 10 axial nodes in the active core channels.

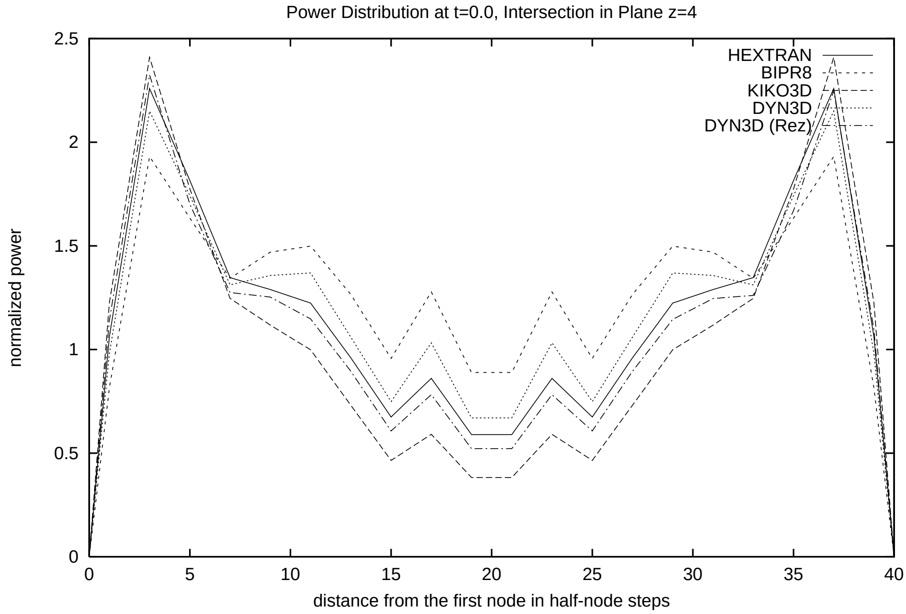


Figure 2. Radial power distribution at an intersection near the core center at steady state of the boron dilution transient.

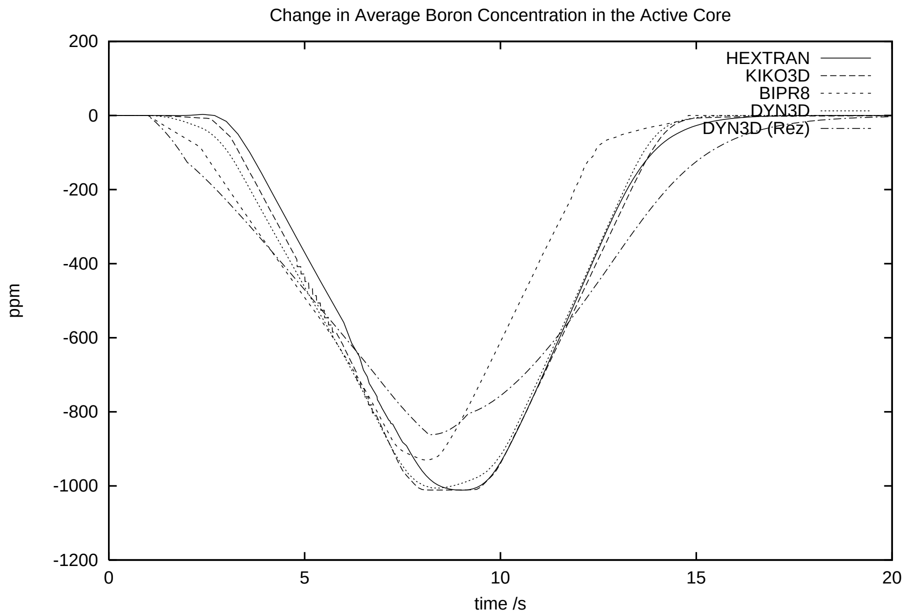


Figure 3. Change in the core average boron concentration during the boron dilution transient.

The prompt overcriticality was achieved between 5.4 - 7.9 s resulting in large fission power peaks. The maximum peaks were from 23000 to 70000 MW. The sensitivity studies with different codes showed that the timing and size of the power peak depends on the amount of numerical diffusion due to spatial or time discretization: more diffusion will decrease the power peak and cause it to occur later. However, the fission power peak of HEXTRAN is largest but it occurs later than in some other results including also the solution with HEXTRAN-PLIM. This is due to the compromise between numerical diffusion and dispersion made in the HEXTRAN calculation: the use of long time steps has minimized the numerical diffusion but induced some dispersion which tends to transfer the minimum of the average boron concentration to a later time. The prompt criticality is achieved later when axially a larger portion of the core is affected by the boron dilution and taking part in the power transient hence causing a larger power increase.

The total power transfered to the coolant is in Figure 4. During the prompt fission power peak the sharp peak due to the gamma power to the coolant can be seen. Also the power integrals and consequently the maximum fuel temperatures differ respectively between the solutions as seen in Figure 5. The axial and radial positions of the maximum fuel temperatures were very close to each other in all results during the transient.

In Figure 6 is the average enthalpy of coolant at the core outlet. The integrated total power is largest in the HEXTRAN solution and consequently the enthalpy increase is largest. However, also the effects of numerical diffusion are interestingly seen in this Figure. Due to numerical diffusion in all other calculations than HEXTRAN the coolant enthalpy decrease disturbance at the inlet of the core is seen at the outlet of the core already before it physically would have travelled there. Due to numerical diffusion the enthalpy decrease is very smooth. In the DYN3D calculation of Øe• this effect is smallest because the minimization of the numerical diffusion of the boron front has been made by using better time and spatial discretizations which affects the tracking of both boron and temperature fronts. In HEXTRAN calculation the numerical diffusion also of the travelling temperature is in control: the first decrease of the outlet enthalpy is physically correctly drowned into the large enthalpy increase due to the fission power peak; later the enthalpy increase back to the original value is

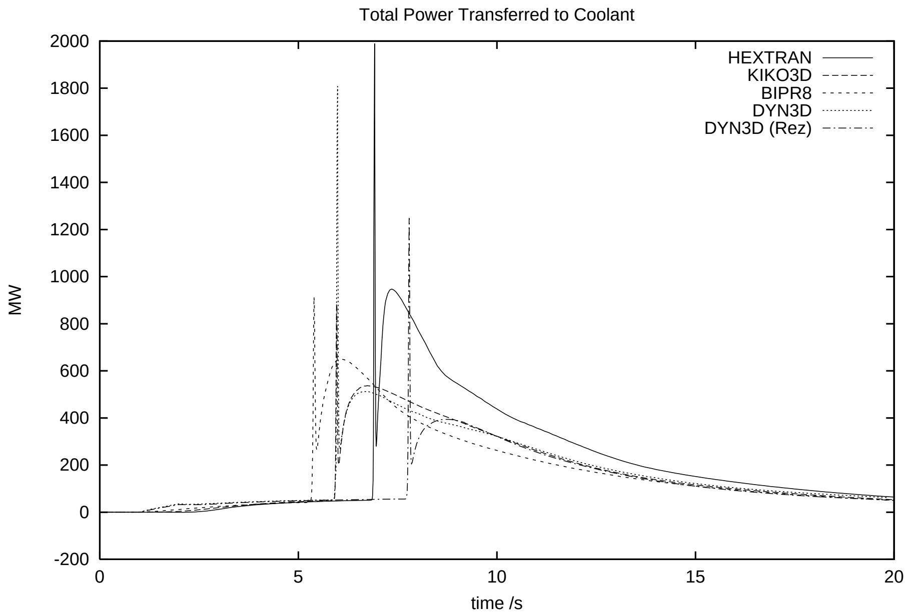


Figure 4. Total power transferred to coolant during the boron dilution transient.

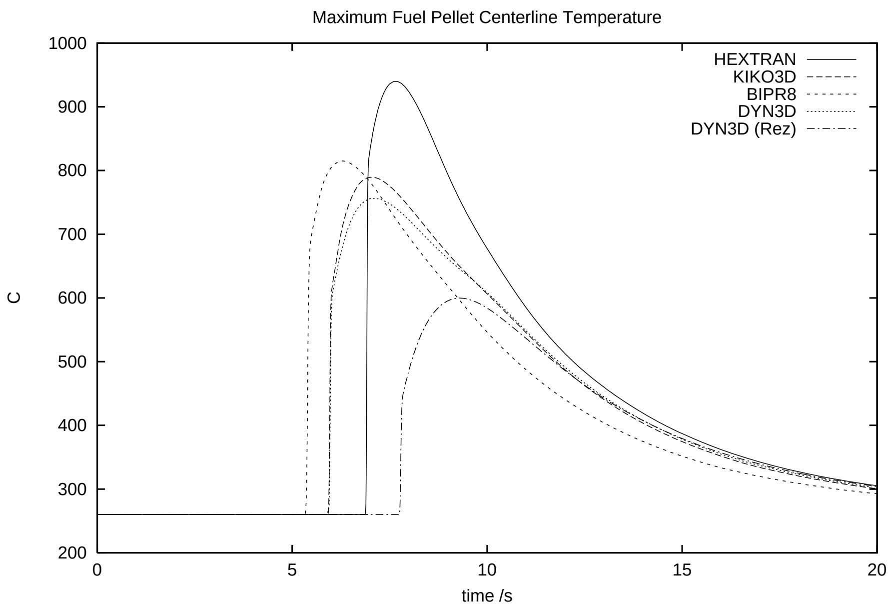


Figure 5. Maximum fuel pellet centerline temperature during the boron dilution transient.

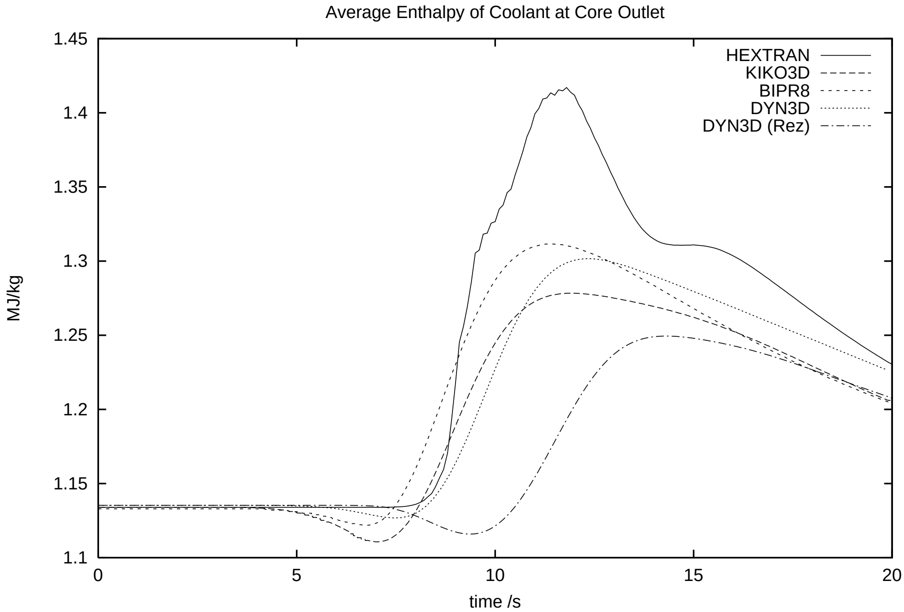


Figure 6. Average enthalpy of coolant at core outlet during the boron dilution transient.

seen in the enthalpy curve contrary to the other calculations where this effect is smoothened out by numerical diffusion.

In this kind of benchmark problem there is no rigorous solution. No clear explanations have been found for the detailed differences in the magnitude and timing of the fission power peaks between the codes. The nuclear data of the fuel and reflectors, power distributions, different thermohydraulics modelling and time discretization methods all affect the results. However, with calculation of this benchmark it has become clear which kind of solution methods and models have to be used so that the solutions are at least approaching the converged solution. In many cases the results could be very nonconservative if normal thermohydraulics methods were used. The results showed a fairly good consistency in spite of the use of the own nuclear data of the participants and the difficulty of the calculational problem in low flow rate conditions. The participants found the benchmark useful in their validation work of three-dimensional reactor dynamics codes.

4. Comparison of HEXTRAN-PLIM and HEXTRAN results for AER's boron dilution benchmark

VTT has long experience in carrying out boron dilution calculations with the HEXTRAN code, and known deficiencies in the existing thermal hydraulic models have been tackled by a proper selection of the space and time discretization and, if necessary, applying overly conservative assumptions. In an earlier natural circulation calculation it was shown that use of conventional methods could lead to severe underconservatism without very conservative assumptions (Rajamäki & Kyrki-Rajamäki, 1997). This can, however, easily lead to overly conservative results.

The benchmark described in the preceding chapter was first calculated with the production version 2.7 of HEXTRAN and then with HEXTRAN-PLIM. In order to avoid numerical diffusion it was taken care in the HEXTRAN 2.7 calculation that the Courant criterium was obeyed for as long as possible; the time step per axial mesh times the flow velocity should be near the value one. Before the fission power peak long time steps of 0.3 seconds were used and the spatial discretization of the heated and unheated parts of the core channels were dense, 25 axial nodes in the heated core (height 2.5 m); the flow velocity was about 0.5 m/s. During the limited duration of the fission power peak shorter time steps had to be used (minimum 0.005 s) but the consequences of this were not dramatic. However, due to long time steps there occurred some numerical dispersion and flow oscillations and conservatively a too large power peak was calculated. The power peak of HEXTRAN-PLIM occurred 0.7 seconds earlier and its integrated power release was some 25 % smaller than in the results of the production version.

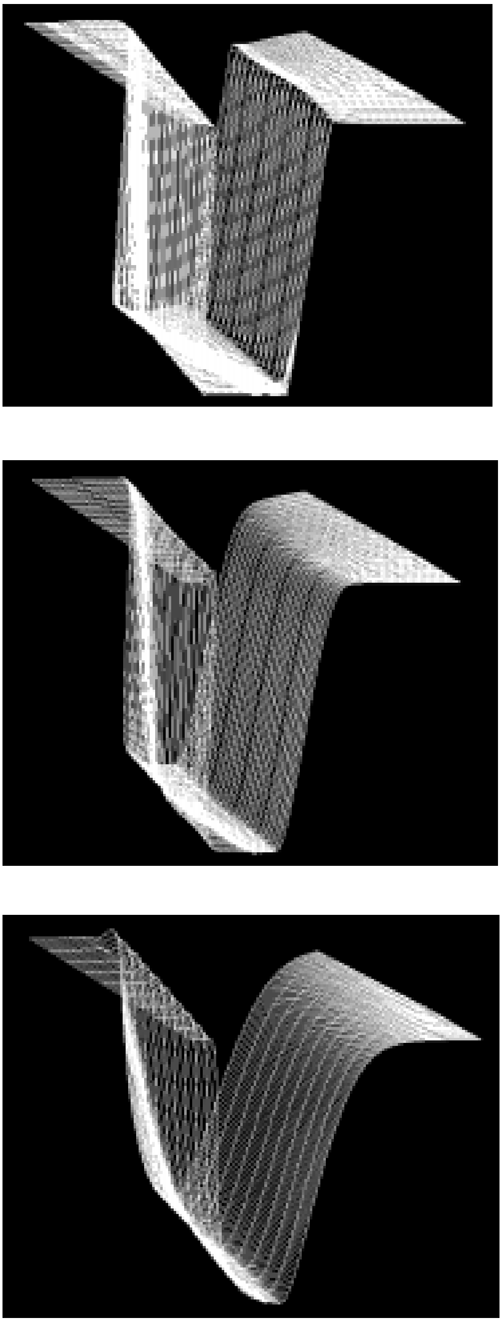
The optimization of time and space discretization described above for the production version of HEXTRAN would be very difficult during more complicated and unpredictable transients. In natural circulation conditions where flow velocity is still much slower the obeying of Courant criterium becomes practically impossible. In HEXTRAN-PLIM the shape preserving capability of PLIM takes automatically care of this kind of problems with any sufficiently dense temporal and spatial meshes.

Demonstration cases have been calculated with HEXTRAN to illustrate the importance of proper modelling, showing examples of possible user effects. In the first demonstration case the number of axial nodes in the core has been decreased from 25 to 10, which is very usual in steady state VVER-440 calculations in Finland, and also in dynamics calculations in other countries where slower codes than HEXTRAN are used. The second demonstration case illustrates the effect of carelessly chosen time-steps in addition to the sparse spatial nodalization: time steps of 0.1 s are used instead of 0.3 s before the fission power peak. The results are catastrophical: the maximum fission power decreases with a factor of 8 both by applying the sparse axial nodalization and again by shortening the time steps; respectively the power peak occurs two and three seconds later.

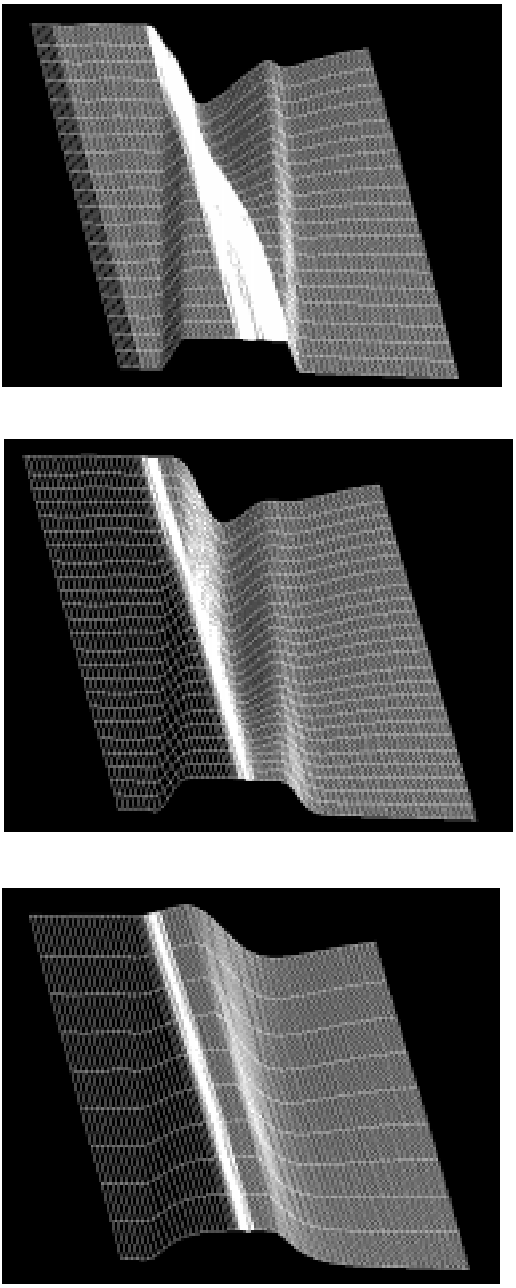
In Figures 7 are the axial distributions of boron density during the transient calculated with HEXTRAN-PLIM, HEXTRAN and the first demonstration case. Growing effects of numerical diffusion and some dispersion are seen, respectively.

In Figures 8 are the axial distributions of coolant density during the transient calculated with HEXTRAN-PLIM, HEXTRAN and the first demonstration case. Growing effects of numerical diffusion are again seen when the modelling is simplified. The travelling of the density disturbance through the core due to the temporary inlet temperature decrease is seen in addition to the density decrease due to coolant heating by the fission power peak.

Use of the PLIM method for this boron dilution benchmark increases the calculation time in the present code version by a factor of 50-150 compared to the production version of HEXTRAN, but no advantage has been taken from parallel computation capabilities of CFDPLIM and this factor is likely to decrease in future code versions. In the one-dimensional TRAB and TRAB-PLIM comparisons computational time when using PLIM increased with a factor of 30-50.



Figures 7. Axial distribution of boron density during the boron dilution transient. Time increases from left to right, and axial position from the bottom upwards. Calculations from the top: HEXTRAN-PLIM, HEXTRAN and demonstration of "user effect" with too few axial nodes in HEXTRAN. Growing effects of numerical diffusion and some dispersion are seen.



Figures 8. Axial distribution of coolant density during the boron dilution transient. Time increases from left to right, and axial position from the bottom upwards. Calculations from the top: HEXTRAN-PLIM, HEXTRAN and demonstration of "user effect" with too few axial nodes in HEXTRAN. Growing effects of numerical diffusion are seen. The travelling of the density disturbance through the core due to the temporary inlet temperature decrease is seen in addition to the density decrease due to coolant heating by the fission power peak.

5. Future development plans

Next step in the CFDPLIM-application will be its extension to circuit models. Development of a BWR circuit model based on the production version of TRAB is in progress. Most features needed in the circuit models - discontinuities, pumps, connecting flow path subregions into flow paths, channel sub-regions in the core, nodes connecting flow paths, closing of the whole circuit and its pressure balance, connecting channels with transverse flow, node-net equations as well as appearing and disappearing of water level - have been tested.

The application of CFDPLIM to the three-dimensional BWR code TRAB-3D will be straightforward based on the experience with the TRAB and HEXTRAN codes, as all the codes have identical hydraulic models in the core.

The long-term goal in VTT's advanced thermal hydraulic modelling is application of the new six-equation model SFAV in reactor dynamics codes in the future. First step in this direction will be applying of SFAV to a simple reactor flow channel.

6. Conclusions

The accurate hydraulic solver CFDPLIM has been applied in VTT's one-dimensional and three-dimensional reactor dynamics codes. The codes of this kind are very complex, since they have to govern a great number of phenomena and couplings. Therefore it is not a simple task to ascertain their reliability and hence their usability. The situation is even more difficult if the errors of numerical methods are allowed to distort known phenomena or to amplify uncertainties. Comparison of results for boron dilution benchmark shows that the codes of different countries yield somewhat scattered results even when aiming at more accurate numerics using partial improvements. Results show nonphysical behavior and overall smoothing, which complicate the interpretation of causes and consequences and hinder the assessment of significance of any computed detail. The results of HEXTRAN-PLIM show very good resolution through the course of the computed transient, eg fronts have undergone different processes but they have conserved as fronts, there are no nonphysical effects, and passing times of advancing effects are correct.

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Validation of the VTT's reactor physics code system

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Abstract

At VTT Energy several international reactor physics codes and nuclear data libraries are used in a variety of applications. The codes and libraries are under constant development and every now and then new updated versions are released, which are taken in use as soon as they have been validated at VTT Energy. The primary aim of the validation is to ensure that the code works properly, and that it can be used correctly. Moreover, the applicability of the codes and libraries are studied in order to establish their advantages and weak points.

The capability of generating program-specific nuclear data for different reactor physics codes starting from the same evaluated data is sometimes of great benefit. VTT Energy has acquired a nuclear data processing system based on the NJOY-94.105 and TRANSX-2.15 processing codes. The validity of the processing system has been demonstrated by generating pointwise (MCNP) and groupwise (ANISN) temperature-dependent cross section sets for the benchmark calculations of the Doppler coefficient of reactivity.

At VTT Energy the KENO-VI three-dimensional Monte Carlo code is used in criticality safety analyses. The KENO-VI code and the 44GROUPNDF5 data library have been validated at VTT Energy against the ZR-6 and LR-0 critical experiments.

Burnup Credit refers to the reduction in reactivity of burned nuclear fuel due to the change in composition during irradiation. VTT Energy has participated in the calculational VVER-440 burnup credit benchmark in order to validate criticality safety calculation tools.

1. Introduction

VTT's reactor physics calculation system includes among others the CASMO (Edenius et al. 1995) fuel assembly burnup code, the MCNP (Briesmeister 1997) and KENO (Hollenbach et al. 1995) Monte Carlo codes, and the ANISN (Parsons 1987), DORT (Rhoades et al. 1988) and TORT (Rhoades et al. 1987) discrete ordinates codes. As a result of increased computer capacity use of the Monte Carlo method in solving the transport equation has increased. The method is flexible. Thus, the geometry and nuclear physics can be modelled rigorously. Nevertheless, deterministic codes will live on as standard tools of reactor physics due to their fastness and ease of use. Besides the codes nuclear data are required for reactor physics calculations. The data applied in the actual calculations are processed by special codes using the so called evaluated data libraries as a starting point. The codes and the data libraries are under constant development and every now and then new updated versions are released, which are taken in use at VTT Energy as soon as they have been validated.

The primary aim of the validation is to ensure that the code works properly and that it is used correctly. Moreover, the applicability of different codes and data are studied in order to establish their advantages and weak points. Critical experiments are used as benchmarks for validation of the reactor physics codes and data libraries. Also, pure calculational comparisons can give valuable information. The fundamental quantities examined in validation are multiplication factors, reaction rates, and power distributions. Experimental benchmarks are used to determine the bias in the calculated multiplication factor. The bias could be due to a faulty calculation method, inaccurate nuclear data, or an error of measurement. Only systematic approach can tell which is the case: VTT Energy is building a sample test library to be routinely used in the future validation.

2. Generation of the program-specific nuclear data libraries from the evaluated data

Evaluated nuclear data should not be taken as given constants of nature but as the best estimates for the nuclear constants. They may introduce inaccuracy in the reactor physics calculations that affects the final results. There are substantial differences between the contents of the program-specific nuclear data libraries incorporated into the different reactor physics code systems. Accordingly, it would be advantageous to be able to generate program-specific nuclear data for different reactor physics codes starting from the same evaluated data. The generation procedure of the program-specific nuclear data libraries is shown in Figure 1. The dotted parts of the procedure have not yet been adopted at VTT Energy. The key player in the procedure is the NJOY (MacFarlane et al. 1994) nuclear data processing system, which is a comprehensive computer code package for producing continuous-energy and multigroup neutron data from the evaluated nuclear data libraries. NJOY has been validated through international cooperation. The VTT's nuclear data processing system is based on the NJOY-94.105 and TRANSX-2.15 (MacFarlane 1992) processing codes.

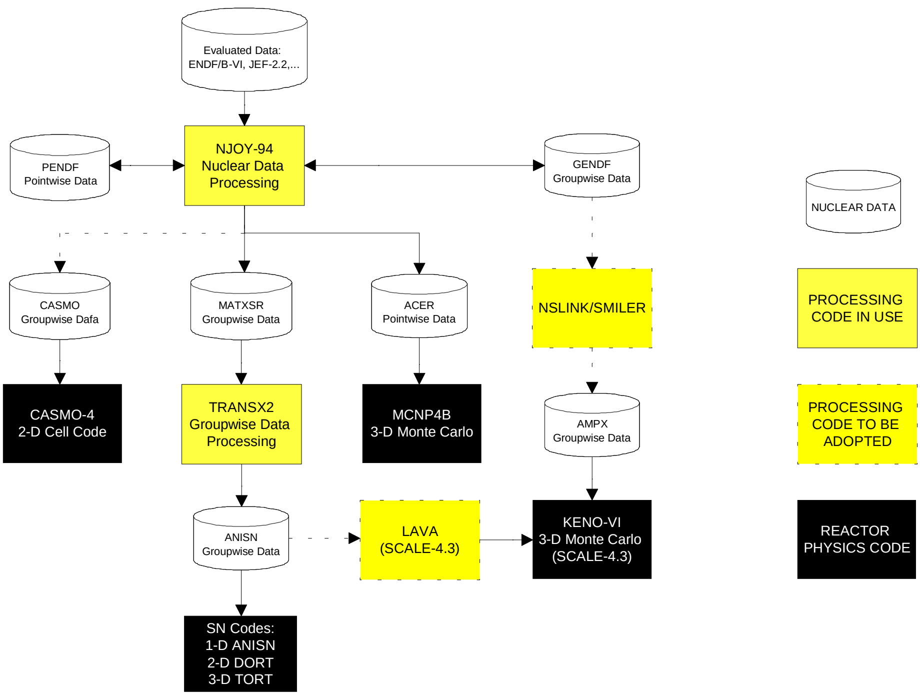

Figure 1. Generation of program-specific data sets from evaluated data.

2.1 Calculation of the Doppler coefficient of reactivity using temperature-dependent program-specific data

NJOY-94.35 and TRANSX-2.15 were applied to generate continuous-energy temperature-dependent data for the MCNP Monte Carlo transport code and groupwise data for the ANISN one-dimensional multigroup discrete ordinates code from the evaluated nuclear data libraries ENDF/B-VI.3 and JEF-2.2 evaluated nuclear data library (Tanskanen 1997a).

The Doppler coefficient of reactivity is a crucial parameter in safety analyses for the transients in light water reactors. The high-temperature program-specific data were applied to perform a set of benchmark calculations for the Doppler coefficient of reactivity for UO_2 (Mosteller et al. 1991) and MOX (Holly et al. 1991) fuel. The results were in a good agreement with the reference results. The calculated Doppler coefficients of reactivity for UO_2 fuel as a function of the fuel enrichment are shown in Figure 2.

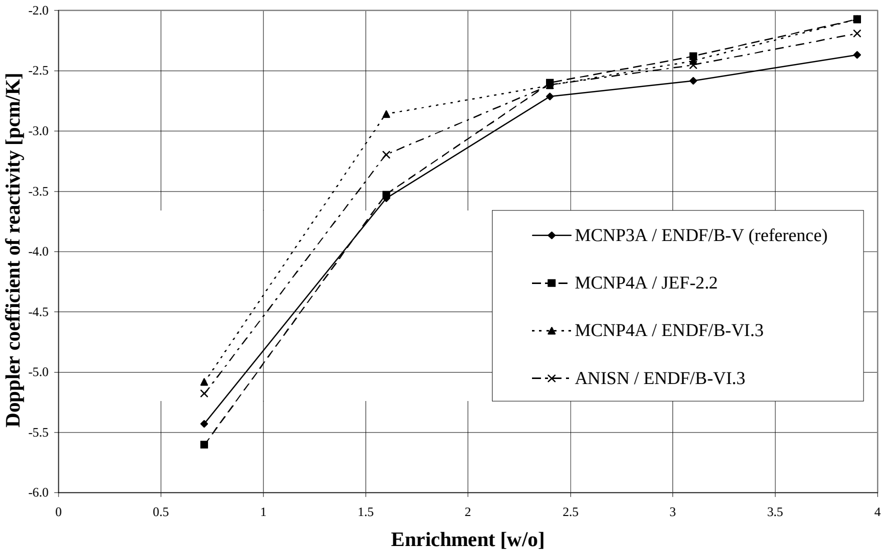

Figure 2. The results of the benchmark calculations for the Doppler coefficient of reactivity for UO_2 fuel calculated using different codes and data.

3. Validation of the KENO-VI code and the 44GROUPNDF5 data library

A new 4.3 version of the SCALE code system was released in 1995. Major enhancements include the addition of KENO-VI Monte Carlo code. KENO-VI is capable of modelling complex geometries, for instance hexagonal arrays. At VTT Energy the KENO-VI code and the 44GROUPNDF5 data library have been validated for VVER applications against ZR-6 and LR-0 experimental data. The latest version of the SCALE code package is SCALE-4.4 that was released in September 1998. VTT Energy has ordered it from NEA Data Bank and it is going to adopted as soon as it has been validated.

3.1 ZR-6 validation calculations

ZR-6 is a Hungarian pool type zero power reactor that has been used to study VVER type fuel (Szatmáry 1997). The experiments are well documented, and suitable for criticality code benchmarking. The hexagonal cores of the ZR-6 experiments were built of fuel pins of length 1.4 m at lattice pitches ranging from 11 to 19 mm, with and without various absorber rods and boric acid in the moderator. The UO_2 fuel was of three enrichments: 1.6, 3.6 and 4.4%. Varying these parameters as well as the core geometry, more than 300 different cores have been studied.

At VTT Energy 29 critical experiments were used to validate the KENO-VI code and the 44GROUPNDF5 data library (Tanskanen 1997b). The experiments included in the validation were of types K91, K271 and X7(Gd). The K91 cores imitate a core consisting of the K91 macrocells containing 91 positions. The size of the K91 macrocells are close to the VVER-440 fuel assembly (VVER-440 fuel assembly has 127 positions). The assembly walls were imitated by a row of water gaps. Accordingly, the K271 cores were built up from the K271 macrocells that resemble the VVER-1000 fuel assembly. The X7(Gd) cores are regular cores were every seventh fuel rod is replaced with a gadolinium rod.

A detailed three-dimensional geometrical model was used in the validation calculations. 200 generations of 1000 neutrons in each generation were simulated for all twelve cases. The bias β (defined as the average calculated multiplication factor minus one) of the calculational method based on the KENO-VI criticality code and the 44GROUPNDF5 data library was found to be -36 pcm. The standard deviation of the calculated multiplication factor was 364 pcm. These validation results can be used to, e.g. establish the sufficient subcriticality of a fuel storage.

3.2 LR-0 validation calculations

LR-0 is a Czech pool type zero power reactor that has been used to study VVER type fuel (Broulik 1991). The experiments carried out at LR-0 include a study of the reactivity of the 19 shortened VVER-440 assemblies with different lattice pitches. The width of the water gap between the assemblies varied from 0.2 to 2.5 cm (3.85 cm in a standard VVER-440 fuel charging pool). The KENO-VI code and the 44GROUPNDF5 data library were validated for criticality safety of fuel charging pool using these twelve LR-0 critical experiments, whose critical water level is shown if Figure 3.

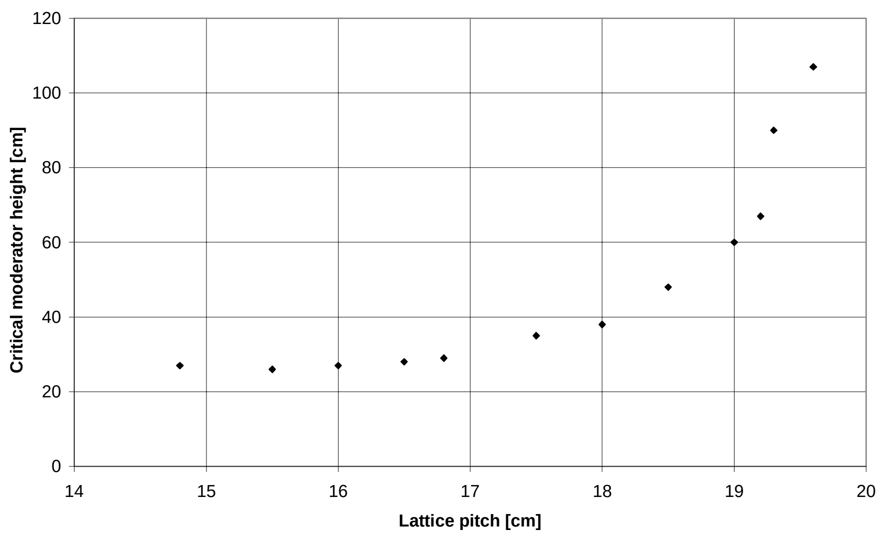

Figure 3. The critical water level as a function of the lattice pitch of the LR-0 experiments.

A detailed three-dimensional geometrical model was used in the validation calculations. The cross section of the calculation geometry in a plane perpendicular to the Z-axis is shown in Figure 4.

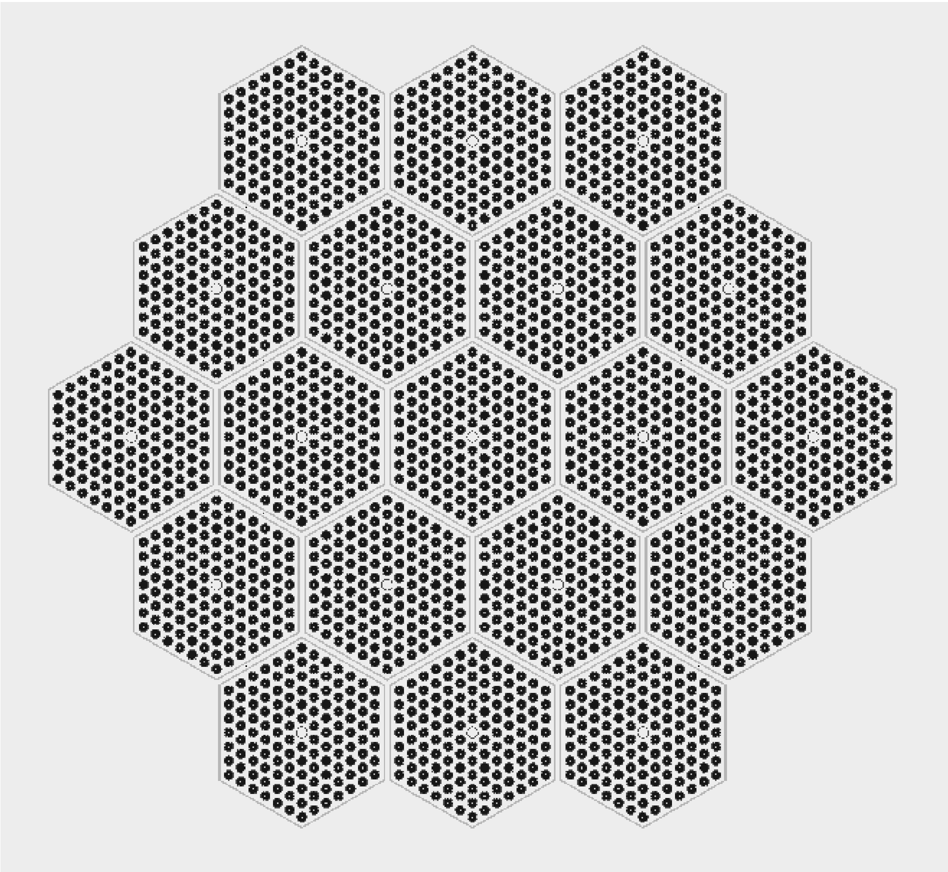

Figure 4. The cross section of the calculation geometry in a plane perpendicular to the Z-axis.

150 generations of 2000 neutrons in each generation were simulated for all twelve cases. First 50 generations were skipped in order to find a realistic source term. The calculated multiplication factors are plotted as a function of the lattice pitch in Figure 5. No significant trend in the calculated multiplication factor as a function of the width of the water gap between the assemblies was encountered. Thus, calculational method based on the KENO-VI code and the

44GROUPNDF5 data library is valid for fuel charging pool criticality calculations.

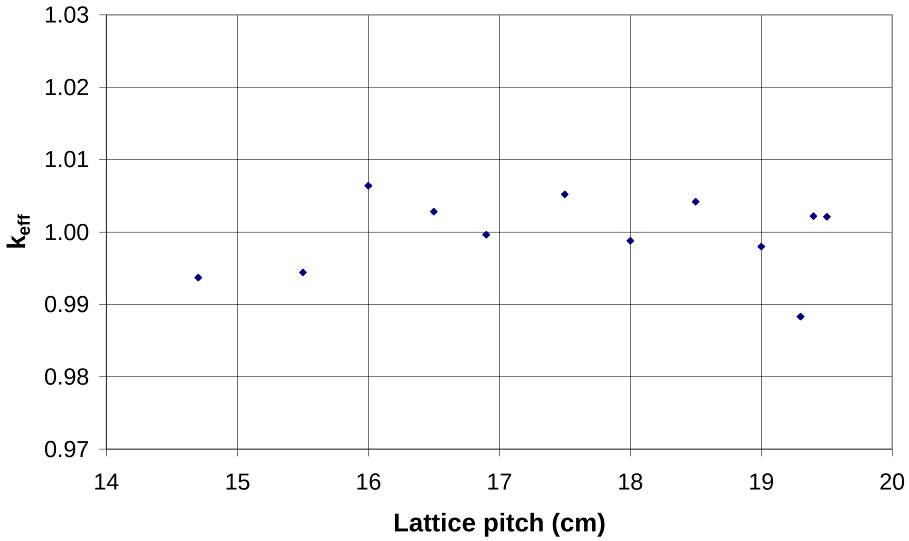

Figure 5. Calculated multiplication factors as a function of the lattice pitch

4. Burnup Credit studies

Burnup Credit refers to the reduction in reactivity of burned nuclear fuel due to the change in isotopic composition during irradiation. The criticality safety analysis of spent fuel has traditionally assumed that the fuel is fresh. This results in excessively large criticality safety margins for typical light water reactor fuel. Use of burnup credit could reduce cost in spent fuel storage and transportation. However, before such an approach can be approved, it is necessary to demonstrate that the calculational tools applied can accurately enough determine the isotopic composition of spent fuel, and predict the subcritical multiplication factor k_{eff} of a spent nuclear fuel package. Burnup Credit approach has been approved in some countries, and many countries are interested in it, because there are indications that nuclear power industry is moving to increased enrichments and higher discharge burnups.

4.1 Calculational VVER-440 burnup credit benchmarks

The first calculational VVER-440 burnup credit benchmark, CB1 (Markova 1996) was proposed on the 6th AER symposium. The aim of the benchmark was to study the effect of 6 major, 5 minor actinides, and 15 major fission products on criticality of the spent fuel.

VTT Energy calculated CB1 using three different code systems: CASMO-4 with E4LBL40, MCNP4B with ENDF60, and KENO-VI with 44GROUPNDF5 (Tanskanen 1997b). The reactivity loss due to the change in isotopic composition during irradiation according to the CB1 results is shown in Figure 6. The burnup of the fuel was 30 MWd/kgU. The major actinides gave about 50% of the reactivity loss. When also the minor actinides and the major fission products were taken into account, about 90% of the reactivity loss was covered. The rest of the reactivity loss is due to the minor fission products, some 200 nuclides. There are different levels of burnup credit method depending on how many nuclides are taken into consideration.

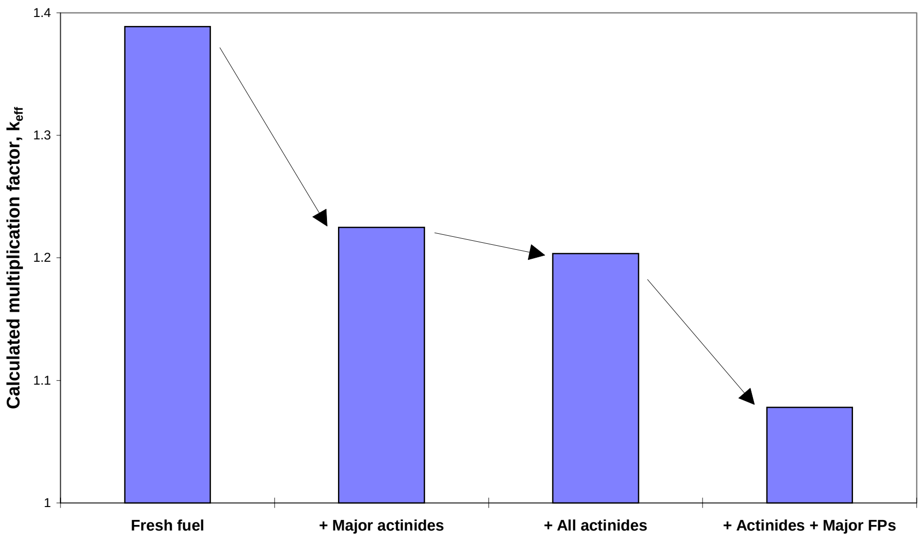

Figure 6. Reactivity loss due to the change in isotopic composition during irradiation according to the CB1 results.

In the second benchmark, CB2 (Markova 1997) the participants were requested to determine the isotopic composition of spent VVER-440 fuel varying burnup and cooling time. At VTT Energy the analysis was carried out using CASMO-4 with E4LBL40 and ORIGEN-S with 44GROUPNDF5 (Tanskanen 1997b). The calculated concentrations of the essential nuclides agreed, but there were some striking discrepancies in the results which were mostly due to the differences in the program-specific nuclear data.

The third benchmark, CB3 (Markova 1998) has also been specified. It is a study of the effect of an axial burnup profile on the multiplication factor. The burnup profiles are given in the specifications of the benchmark. The participants are requested to compare the reactivity of a fuel rod with a burnup profile with that of a fuel rod with a flat burnup profile. VTT Energy intends to calculate the benchmark using the MCNP4B and KENO-VI three-dimensional Monte Carlo codes.

5. Summary and conclusions

At VTT Energy several international reactor physics codes and nuclear data libraries are routinely used. New updated versions of the codes and data are adopted as soon as they have been validated at VTT Energy. The primary aim of the validation is to ensure that the code works properly, and that it can be used correctly. The second aim is to establish the advantages and the weak points of different codes and data.

VTT Energy has acquired a validated nuclear data processing system based on the NJOY-94.105 and TRANSX-2.15 processing codes. The validity of the system has been demonstrated by generating pointwise (MCNP) and groupwise (ANISN) temperature-dependent cross section sets for the benchmark calculations of the Doppler coefficient of reactivity. The future aim is to be able to generate program-specific data for any reactor physics code applied at VTT Energy.

As a result of increased computer capacity use of the Monte Carlo method in solving the transport equation has increased enormously. The flexibility of the method enables rigorous modelling of geometry and nuclear physics.

Nevertheless, deterministic codes will live on as standard tools of reactor physics thanks to their fastness and ease of use. At VTT Energy an increasing attention will be paid to Monte Carlo method in the future. One subject to be studied is use of Monte Carlo technique in burnup calculations.

At VTT Energy the KENO-VI Monte Carlo code is applied to criticality safety analyses. The calculation method based on the KENO-VI code and the 44GROUPNDF5 data library has been validated at VTT Energy against the ZR-6 and LR-0 critical experiments. The code and its data library have turned out to be competent to criticality safety analyses.

Burnup Credit refers to the reduction in reactivity of burned nuclear fuel due to the change in composition during irradiation. VTT Energy has participated in the calculational VVER-440 burnup credit benchmarks. The CB1 benchmark clearly demonstrated the potential of burnup credit. In addition, the benchmarks have served as validation of the VTT's reactor physics codes.

The validation described in this report has been aimed at the ability of the code systems to predict the multiplication factor of experimental lattices. However, multiplication factor is an integral quantity which may not imply the underlying cause of the biased result. Further, errors could cancel each other out. Hence, in the future priority will be given to detailed studies of reaction rates and power distributions.

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Thermal hydraulic experiments for VVER reactor safety

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Abstract

TEKOJA project concentrated on thermal hydraulic experiments for VVER reactor safety. The backbone of TEKOJA has been tests using a PACTEL test loop. PACTEL is volumetrically scaled out-of-pile model of VVER-440 reactors operated in Finland. Results of two investigations has been highlighted in this paper. In a natural circulation study, primary coolant loop flow dependence on pressure was shown. This has relevance in reactor safety studies when the natural circulation with reduced inventories are considered. In an ATWS examination, shifting of flow between primary loops was measured in a PACTEL test. This finding supported safety analysis computer code calculations for a loss-of-feedwater incident. The initial reason for the periodic change of the loop was deteriorating heat transfer in an active loop, which changed the distribution of the coolant in the primary side.

1. Introduction

TEKOJA project concentrated on the thermal hydraulic reactor safety experiments for VVER type pressurised water reactors used in Finland. The backbone of TEKOJA has been tests using a PACTEL test loop. PACTEL is the largest and most versatile integral test facility of VVER type reactors in the world. The conducted tests produced data for validation of safety analyses computer codes, supported development of accident management measures and provided information about safety system performance during abnormal operation of a reactor.

This paper highlights results of two studies carried out in TEKOJA project with PACTEL. An investigation on VVER natural circulation has been performed and the outcome of this study is presented here. The other outstanding result chosen for this paper was findings of a loss-of-feedwater ATWS experiment.

This report consists of a short description of the PACTEL test loop in section two. Section three presents the results of the natural circulation investigation and section four shows the findings of the ATWS test.

2. Description of the PACTEL test loop

PACTEL is volumetrically scaled (1:305) out-of-pile model of VVER-440 reactors operated by Imatran Voima Oy and located at Loviisa, Finland. The test loop includes all the main components of the reference reactor: a pressure vessel, hot and cold legs, steam generators and a pressurizer. In addition to these low and high pressure emergency core cooling systems are simulated. The maximum operating pressure on the primary side is 8.0 MPa and 5.0 MPa on the secondary side. The pressure vessel is modelled with a U-tube construction consisting of separate downcomer and core sections. The elevations are the same as in the reference reactor in order to maintain the pressure heads of reference reactor during natural circulation. Figure 1 shows the main components of PACTEL.

The core is comprised of 144 electrically heated fuel rod simulators. The peak core heating power is 1000 kW, which is about 22 % of the scaled down nominal power (1375 MW in reference power plant). The rods are divided into three parallel channels and each channel represents part of one fuel bundle.

Six coolant loops of the VVER-440 are simulated with three loops in PACTEL. A heat transfer area of a PACTEL steam generator corresponds to two scaled down reference steam generators. Almost horizontal position and U-tube construction of the reference system has been reproduced in the heat exchange tubes. There are 118 tubes in 14 layers and tube diameter is the same as in the power plant.

PACTEL measurement instrumentation consists of about 550 transducers. Altogether 450 thermocouples and RDT transducers are used for temperature measurement. Differential pressure transducers are used for liquid level, void fraction and pressure loss measurements (68 transducers). Mass flow rate is measured in the cold legs and in the downcomer by a wide range vortex flow meters. Pressure is measured at several locations both in primary and secondary side. Core heating power is determined with Norma power analyser. Hewlett-Packard HP 3852A is used as a data acquisition unit. The PACTEL test facility is operated by VTT Energy and it is located in a nuclear engineering laboratory of Lappeenranta University of Technology. More detailed PACTEL description is presented in reference (Tuunanen et al. 1998).

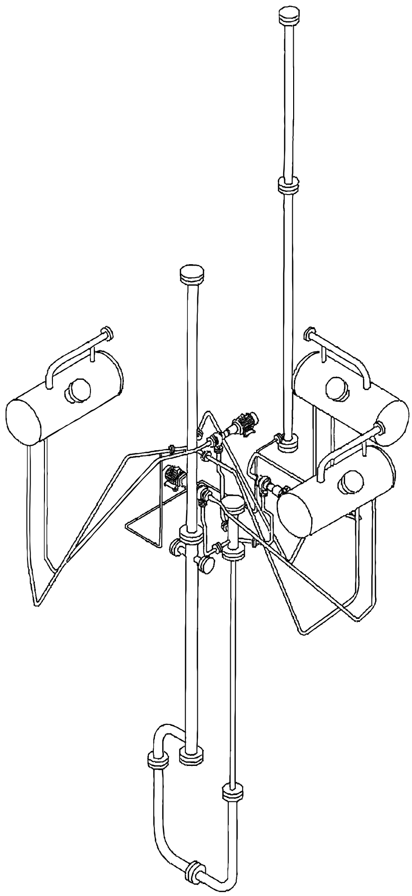

Figure 1. The PACTEL test loop.

3. Natural circulation experiments

Natural circulation flow develops in a closed loop in which a heat source is located at a lower elevation than a heat sink. Mass flow rate in the loop depends on density difference between the heat source and the heat sink, the difference in elevation, and frictional losses in the loop. Natural circulation is utilised in nuclear reactors both in normal operations and during postulated accident situations. In fact, primary coolant pumps are stopped in most of the abnormal events. That is why natural circulation is one of the key phenomenon in many reactor safety analyses. A large data base, which is available for PWRs can not be applied for VVER reactors as such, because of the different loop geometry. This has been the motivation for the natural circulation studies carried out in national reactor safety research programmes.

VTT Energy has performed an extensive test programme on natural circulation in VVER reactor geometry. The experiments started in 1985 by using a REWET-III test facility (low pressure) and continued with the PACTEL facility in 1990's. The natural circulation tests include a wide spectrum of experiments: stepwise inventory reduction experiments, small break LOCA tests, hot leg loop seal tests and secondary side loss-of-feedwater experiments. The effect of primary pressure on characteristics of natural circulation and hot leg loop seal behaviour was studied during TEKOJA project. The findings of these natural circulation tests are presented here.

Three experiments were performed to reveal the effect of primary pressure on characteristics of natural circulation (SIR-20, SIR-21, SIR-23). The primary inventory was reduced in steps and only one of the three loops was used in the tests. The main difference in the experiments was the initial primary and secondary pressure. The pressurizer was isolated in each test to make the determination of inventory simpler. Roughly 40 kg of the primary coolant was drained through a lower plenum drain valve in each step. After each outlet the loop was allowed to stabilise for 1000 seconds. There was an exception in SIR-23 experiment, where one draining was started earlier in order to avoid the pressure exceeding the facility maximum pressure. Table 1 shows the initial conditions for each experiment.

Table 1. Initial conditions for SIR-23, SIR-20, and SIR-21experiments.

	SIR-23	SIR-20	SIR-21
Primary pressure (MPa)	**7.5**	**4.0**	**1.6**
Secondary pressure (MPa)	**4.2**	**1.2**	**0.3**
Core Power (kW)	**115**	**115**	**115**

Figure 2 presents the mass flow rate of the loop as a function of time in two tests SIR-21 and SIR-23. Three different heat transfer modes were measured in the experiments. Single-phase natural circulation was established when the primary circulation pumps were halted at 1000 seconds. Temperature in the upper plenum and in the hot leg was subcooled. The upper plenum and the hot leg conditions changed from subcooled to saturated as a consequence of the first depletion, but the loop flow was still single-phase flow. After the second depletion (at 3000 s) the collapsed upper plenum level dropped close to the hot leg entrance elevation. In the high pressure test the heat transfer started to deteriorate rapidly, because of decreasing loop flow. This initiated a pressure peak, and the operators opened the drain valve about 500 seconds earlier than planned, before the pressure exceeded the maximum operating pressure of the test facility. The loop flow started to decrease in the low pressure experiments as well, but mass flow rate declined very slowly. It is possible that if longer period between depletions were applied, flow stagnation would have eventually occurred.

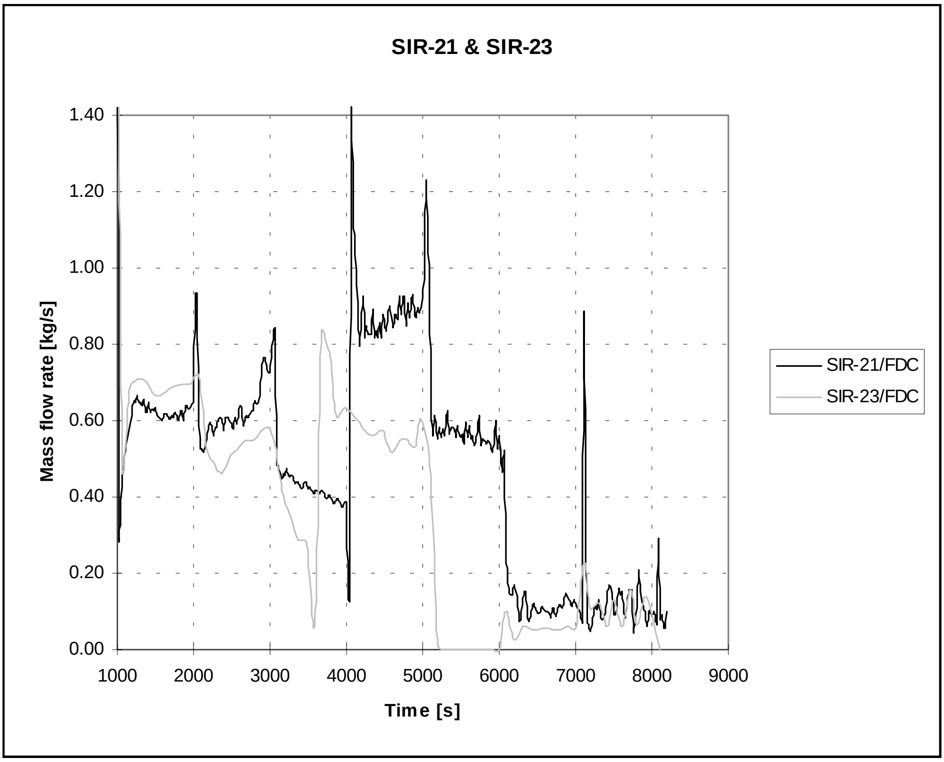

Figure 2. Downcomer mass flow rate as a function of time in SIR-21 and SIR-23. Uncertainty is ± 0.03 kg/s and readings are out of range after 6000 seconds.

When the third drainage was concluded the flow regime changed from single-phase to two-phase flow. The loop mass flow rate increased in two low pressure tests when compared to the single-phase mass flow rate. In high pressure test the two-phase flow mass flow rate remained clearly at a lower value than what was observed in single-phase flow. After the fourth depletion two-phase heat transfer mode continued in low pressure experiments while boiler-condenser natural circulation flow was measured in SIR-23. The transition from two-phase flow to boiler-condenser flow occurred after the fifth outlet in low pressure. The heat transfer from primary side to secondary side continued in boiler-condenser mode until the end of each experiment.

The behaviour which has been observed earlier when REWET-III and PACTEL tests were compared, was duplicated in this test series (Kervinen & Hongisto 1986 and Lomperski & Kouhia 1994). The loop flow characteristic was depending on the pressure. When the primary pressure was high, the maximum loop flow was observed in single-phase natural circulation flow (SIR-23), where as in low pressure conditions the maximum flow was measured in two-phase flow regime (SIR-20, SIR-21). The explanation for this behaviour is the change in steam density when the pressure changes. Steam density is very low at low pressure (large volume) and the level swell is higher in the upper plenum. More water is flowing from the upper plenum to the hot leg than in the high pressure case. This also means that the flow stagnation occurs at a lower inventory in low pressure than in the high pressure. In addition to this a transition from two-phase natural circulation to boiler-condenser natural circulation occurs at lower inventory in the low pressure. The core heat up occurs at lower inventory in low pressure than in high pressure. Figure 3 shows the normalised loop mass flow rates as a function of primary side inventory. The mass flow rate with 100 % inventory has been used to calculate dimensionless mass flow rate for each test.

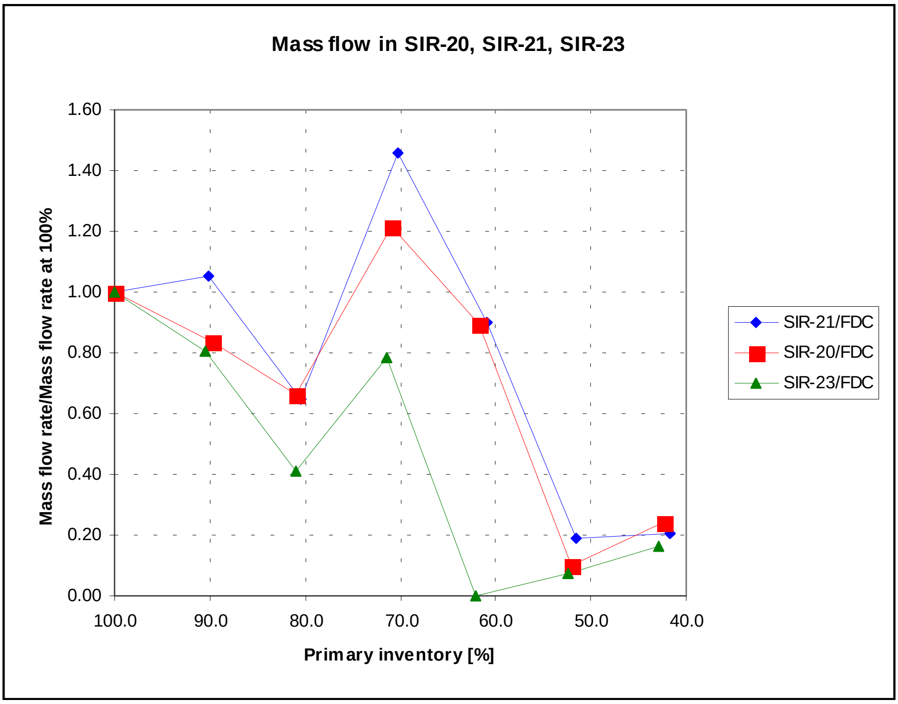

Figure 3. Loop mass flow rate as a function of primary inventory. Mass flow rate is normalised to the single-phase flow. Readings are not valid at inventories less than 60 %. Lines were added to the graph for clarity.

The findings of this study can be applied in small break loss-of-coolant accident analyses for the nuclear power plants. One aspect of such a transient is possible boron dilution during boiler-condenser heat transfer mode as Hyvärinen (1993) has demonstrated. In Loviisa power plant the operators reduce the primary pressure by cooling down the secondary side, if loss-of-coolant accident is detected. The reason for this action is to ensure that primary pressure will decrease below delivery head lift of low pressure safety injection pumps. This means that inventory changes are occurring at lower pressure than the operating pressure of the secondary side. The safety analysis computer codes should predict well the natural circulation mode transitions as a function of pressure in order to provide adequate information for boron dilution studies.

The observed loop flow characteristic is prototypical for the VVER-440 geometry. However, the measured characteristics can not be applied to multiloop system as such, since the loops can be in different natural circulation mode at the same time. In fact, this has been the case in most multiloop natural circulation experiments performed with PACTEL. The pressurizer which was disconnected in this test series has a significant effect on the loop flow. The coolant surging in and out of the pressurizer changes the inventory in the rest of the primary circuit. Together with hot leg loop seal the pressurizer can cause strong oscillations in mass flow rate. An exhaustive report on the SIR-20, SIR-21 and SIR-23 experiments is given by Semken & Tuunanen (1995).

4. ATWS experiment

When safety analysis for a nuclear power plant are being carried out, a failure of reactor scram together with different initial events, have to be considered. This type of incidents are called Anticipated Transient Without Scram (ATWS). One such an event is a loss-of-feedwater transient as an ATWS case. In the safety analysis for Loviisa power plant the primary loop, which took care most of the heat transfer, changed periodically during the late phase of the simulation. The aim of the investigation was to study flow shifting from one loop to another during similar ATWS transient. The main objectives were to find out, if it was possible to duplicate shifting of the loop flow in a test, and to explain the mechanism which caused the cyclic behaviour. In addition to those the experiment demonstrated the use of the PACTEL test loop for transients in which the parameters are beyond the test facility's design criteria.

Two experiments ATWS-5 and ATWS-6 were performed in this study. The ATWS-5 test helped in finding correct parameters for the actual experiment ATWS-6. The feedwater mass flow rate for each steam generator was 0.02 kg/s. This flow rate corresponds to the capacity of one emergency secondary side feedwater pump in the reference power plant. Core power was selected so that the feedwater flow rate was not sufficient to maintain the level in the secondary side. The secondary side pressure was 3.0 MPa through out the test. The initial values of main parameters in ATWS-6 are presented in Table 2.

Table 2. Initial main parameter values for ATWS-6.

Primary pressure	**7.0 MPa**
Secondary pressure	**3.0 MPa**
Core heating power	**260 kW (5.8%)**
Secondary level	**20 cm**
Emergency feedwater mass flow rate	**0.02 kg/s / SG**
Primary loop mass flow rate	**2.0 kg/s / loop**
Opening pressure of safety valve	**7.5 MPa**
Closing pressure of safety valve	**7.3 MPa**
Safety valve orifice diameter	**4 mm**

Figure 4 shows the behaviour of the primary pressure in ATWS-6. The experiment began with the system stabilised at 100 % primary coolant inventory with the main circulation pumps running. After 1000 s of steady state, the main circulation pumps started to coast down. The primary pressure rose, because the core power was higher than the steam generators and the heat losses could remove. The primary pressure rose until the opening pressure (7.5 MPa) of the pressurizer safety valve simulator was reached at roughly 7000 seconds. Then the primary coolant inventory decreased through the repeatedly opening and closing safety valve. The pressurizer safety valve cycled on and off up until 11300 s. During that time about 160 kg of the coolant was lost.

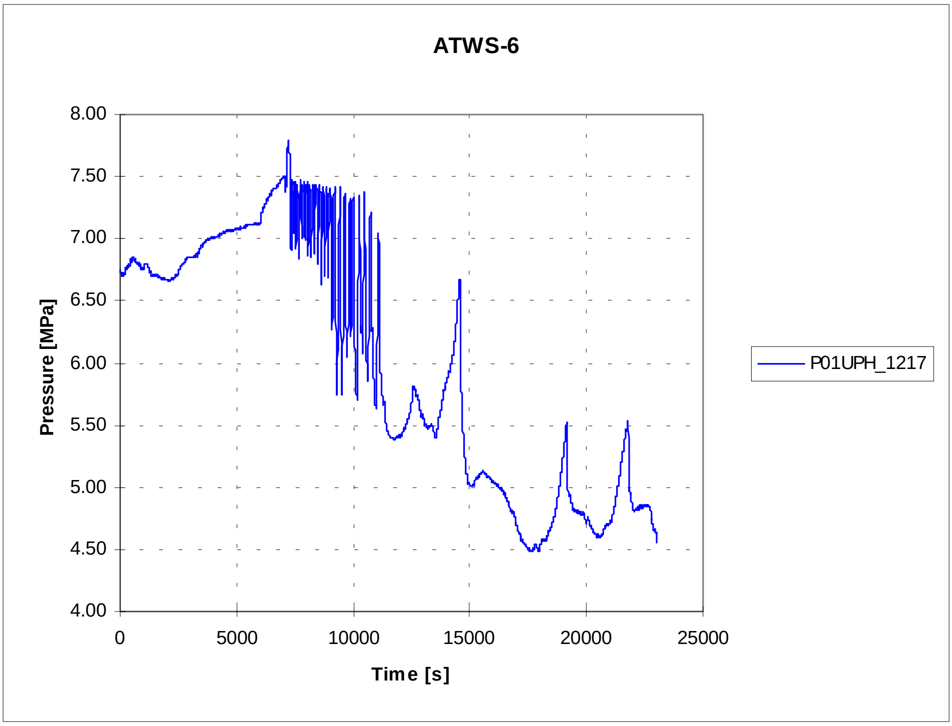

Figure 4. Primary pressure history in ATWS-6 (Uncertainty ± 0.03 MPa).

When the cyclic operation of the pressurizer safety valve stopped, the bulk of the heat transfer was through loop 3. Because the hot leg loop seals hampered loop flow into steam generators 1 and 2, the secondary side feedwater injection increased the secondary side collapsed levels in those steam generators. The loop seal of loop 3 was clear, and steam entered the steam generator. The secondary side feedwater injection into steam generator 3 could not compensate for the evaporation rate in the secondary side, and the secondary side level decreased in steam generator 3 (Figure 5).

When the secondary side swell level in loop 3 was below the bottom of the fourth heat exchanging tube layer, the wetted heat transfer area decreased about 33 %. The heat transferred from the core to the steam generators was not sufficient anymore, and boiling became more intense in the core, which initiated rise in pressure. Primary coolant began to flow to the pressurizer and the upper plenum level dropped slowly. To maintain the manometric balance, the down-flow side of the hot leg in loops 1 and 2 dropped slowly towards the bottom of the loop seal.

When the next heat exchanging tube layer began to uncover, the flow stagnated, and the primary pressure rose abruptly. The tube layer represents about 1/2 of the heat exchange area that was still under water. As the water level reached the bottom of the loop seals, the loop seals cleared, and steam flowed up towards the steam generators. When steam filled the up-flow side of the hot leg, the manometric balance between the cold and the hot side was lost, and the resulting pressure imbalance forced water into the core. The resumption of flow lowered the primary pressure as steam was condensed in the steam generators, and the accumulated coolant surged from the pressurizer. The resumed flow restored the pressure balance around the loop so the flow rate soon dropped.

When the coolant returned from the pressurizer, the primary coolant inventory was the same as before the deterioration of the heat transfer started. So, coolant filled the same number of hot leg loop seals as before, but this time loop 2 transferred most of the energy. Hot leg loop seal 1 refilled, because the pressurizer is connected to the beginning of that hot leg. This time loop 2 had higher pressure head in the cold side than what loop 3 had, because there was higher density liquid in the loop 2 cold leg. So, the loop flow shifted to loop 2. The cycle was repeated with flow shifting between loops 2 and 3. The interval between the loop change became shorter each time, because the secondary side level was gradually decreasing in steam generators 2 and 3.

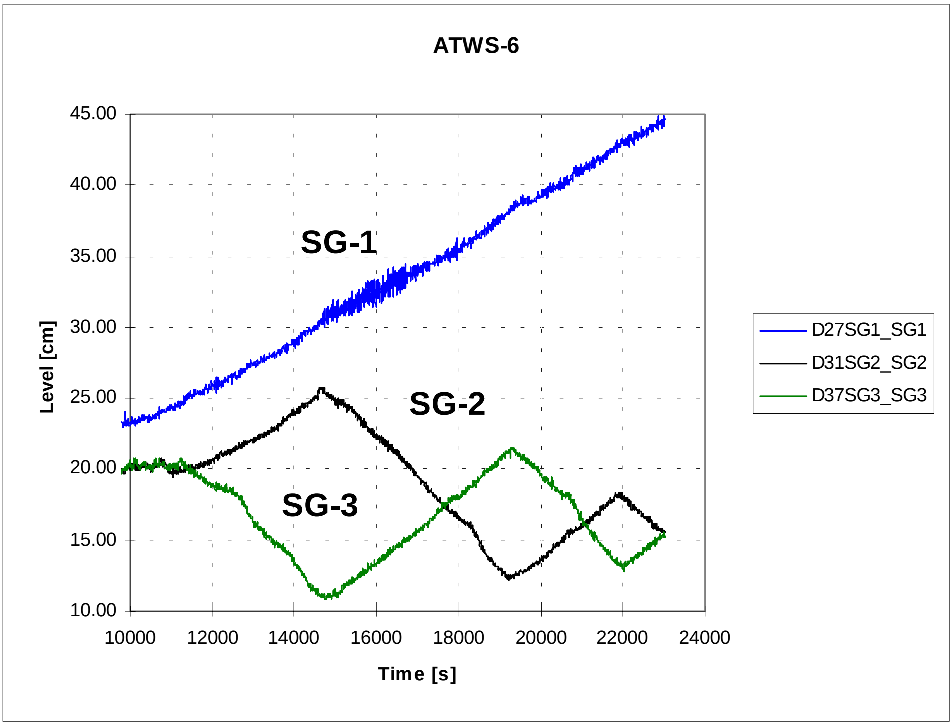

Figure 5. Steam generator secondary side level as a function of time in ATWS-6 Uncertainty ± 1.5 cm). Notice, that time scale is 10,000 - 24,000 seconds for clarity.

Figure 5 presents the steam generator secondary side levels during the last 13,000 seconds of the test. The level measurements indicated clearly how level was decreasing in one loop at a time while the level was increasing in the other two loops at the same time. Level was increasing in steam generator 1 during the flow shifting period, which pointed out that there was very little heat transfer occurring in loop 1. The steam generator level behaviour is the most illustrative indication of the loop flow shift during the test.

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Passive safety injection experiments and analyses (PAHKO)

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Abstract

PAHKO project involved experiments on the PACTEL facility and computer simulations of selected experiments. The experiments focuced on the performance of Passive Safety Injection Systems (PSIS) of Advanced Light Water Reactors (ALWRs) in Small Break Loss-Of-Coolant Accident (SBLOCA) conditions. The PSIS concisted of a Core Make-up Tank (CMT) and two pipelines (Pressure Balancing Line, PBL, and Injection Line, IL). The examined PSIS worked efficiently in SBLOCAs although the flow through the PSIS stopped temporarely if the break was very small and the hot water filled the CMT. The experiments demonstrated the importance of the flow distributor in the CMT to limit rapid condensation. The project included validation of three thermal-hydraulic computer codes (APROS, CATHARE and RELAP5). The analyses showed the codes are capable to simulate the overall behaviour of the transients. The detailed analyses of the results showed some models in the codes still need improvements. Especially, further development of models for thermal stratification, condensation and natural circulation flow with small driving forces would be necessary for accurate simulation of the PSIS phenomena.

1. PACTEL EXPERIMENTS

1.1 Experiment parameters and procedure

The PACTEL experiment programme included three series with altogether 15 experiments (Tuunanen et.al, 1998). See Table 1. In the experiments, all three loops of the PACTEL were in operation. The first series focussed on break size effects on the PSIS behaviour, the second concentrated on studies of break location influences and the third series studied influences of the CMT position on the PSIS behaviour. The main interest in all experiments was in the PSIS behaviour. The main phenomena of interest were the PSIS flow rate, heat transfer to the CMT walls and thermal stratification and condensation in the CMT. The CMT was the only safety injection system in use. The experiments did not include accident management procedures, such as depressurization of the primary or secondary circuits, which are a vital part of the AP600 accident management procedures.

Table 1. Main parameters of the PACTEL experiments.

EXPERIMENT	**BREAK DIAMETER (mm)**	**BREAK LOCATION**	**CORE POWER (kW)**	**OBJECTIVES**
First series				
GDE-21	1	cold leg close to DC	160	break size effects
GDE-22	2,5	cold leg close to DC	160	break size effects
GDE-23	5	cold leg close to DC	160	break size effects
GDE-24	3,5	cold leg close to DC	160	break size effects
GDE-25	3,5	cold leg close to DC	160	reproducibility of the phenomena
Second series				
GDE-31	3,5	cold leg close to DC	160	small CMT
GDE-32	3,5	hot leg loop seal	160	break location, flow reversal in cold leg
GDE-33	3,5	cold leg between steam generator and pressure balancing line	160	break location, flow reversal in cold leg
GDE-34	3,5	cold leg close to DC	160	hot CMT; no recirculation flow
GDE-35	3,5	cold leg close to DC	160	no sparger; condensation in the CMT
Third series				
GDE-41	3,5	cold leg close to DC	160	CMT position; increased driving force for CMT flow
GDE-42	3,5	cold leg close to DC	160	additional IL flow orifice
GDE-43	1	cold leg close to DC	160	long recirculation phase; disappearance of driving force for injection
GDE-44	3,5	cold leg close to DC	160	cold CMT; PBL heating
GDE-45	3,5	cold leg close to DC	160	PBL connected to PRZ

The experiment procedure was similar in all experiments. To prepare for the experiments, the PACTEL operators filled the primary and secondary systems with water, and heated the loop to the desired initial conditions. After reaching the desired initial conditions, the operators kept it for one hour before beginning the experiment. The initial conditions included a steady-state single-phase forced circulation in the primary loops. The initial primary and secondary pressures were about 4.3 and 2.0 MPa. The pressurizer heaters controlled the primary pressure and the secondary side controller the secondary pressure. The core power was 160 kW. The operators used the feed water pumps manually to keep the secondary side level above the heat exchange tubes. In the PSIS, all the valves in the PBL were open all the time. The IL check valve was closed during the heat-up and steady state phases of the experiments.

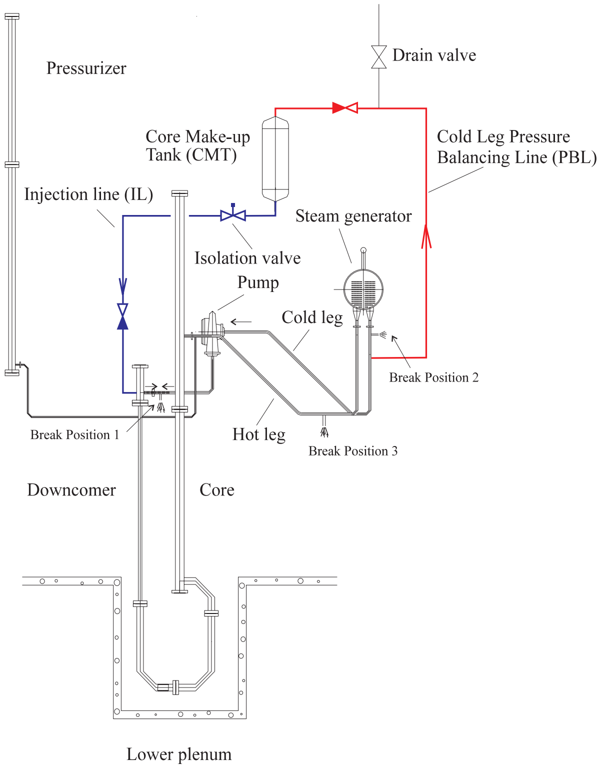

Figure 1. Principle configuration of PACTEL in the passive safety injection experiments.

The first 1000 seconds of the experiments included steady-state measurements. Before opening the break, the operators opened the PBL drain valve to fill the line with hot water. The transient began at 1000 seconds when the operators opened the valve downstream the break orifice and stopped the primary circulation pumps. The operators opened the IL isolation valve and switched off the pressurizer heaters when the pressurizer level dropped below 3,5 metres. The operators finished the experiments when the surface temperatures of the fuel rod simulators at the core exit region exceeded 300 ^{o}C, or when the experiment had lasted 15000 seconds.

1.2 PSIS configurations

The PSIS configuration varied in different series. Figure 1 presents the principle PSIS configuration. The PSIS consisted of PBL, CMT and IL. The PBL connected the CMT to the cold leg. The IL connected the bottom of the CMT to the downcomer. The IL isolation valve was closed during normal operation of the loop. The PBL was open all the time and the CMT was at same pressure as the primary circuit. The CMT instrumentation included thermocouples in different elevations for water and wall temperature measurements, differential pressure transducers for level measurements and flow meaters in the PBL and IL for detection of single-phase liquid flow in the lines.

1.3 Results

1.3.1 PSIS operation modes

The flow through the PSIS began when the operators opened the IL isolation valve from the low pressurizer level. The first PSIS operation mode, *recirculation phase*, included single-phase water circulation through the system. The density difference between the hot water in the PBL and the cold in the CMT and IL created the driving force for the flow. The second mode, *oscillating phase*, included two-phase flow in the PBL. The density difference which drove the flow was larger since the PBL was now full of two-phase mixture. The flow rate during this phase was larger than during recirculation phase. The third operation mode was called *injection phase*. During this phase, steam flowed to the CMT and level in the tank dropped. The driving force for flow was large and the flow through the PSIS was at it's maximum. See Figure 2 and 3. The recirculation phase was important, because the hot water flowing to the CMT formed an isolating layer in the tank between the cold water and steam. This reduced possibilities for rapid condensation in the CMT. Condensation may disturb PSIS operation, if the hot liquid layer breaks down. This may happen, for example, during oscillating phase when two-phase mixture flows to the CMT.

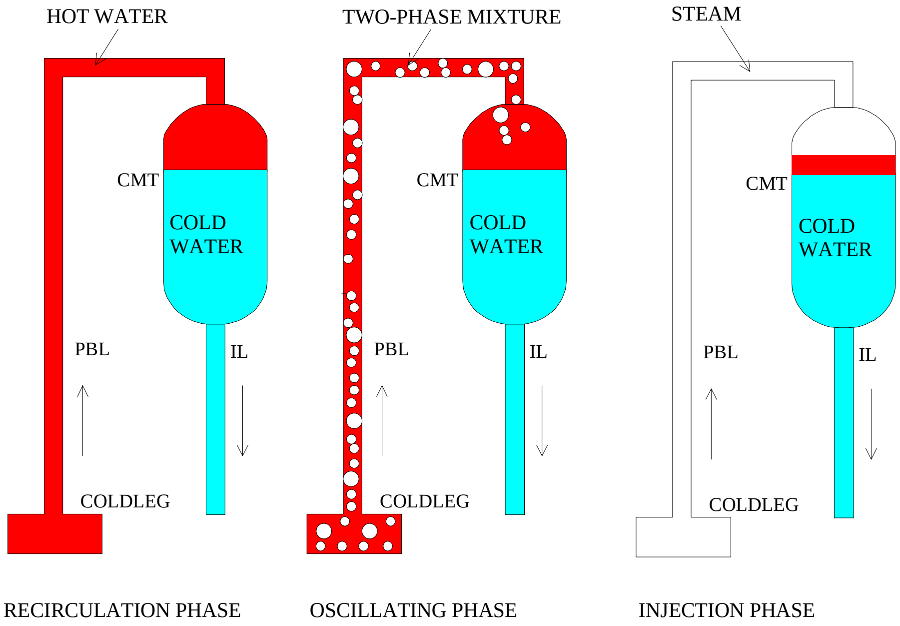

Figure 2. PSIS operation modes.

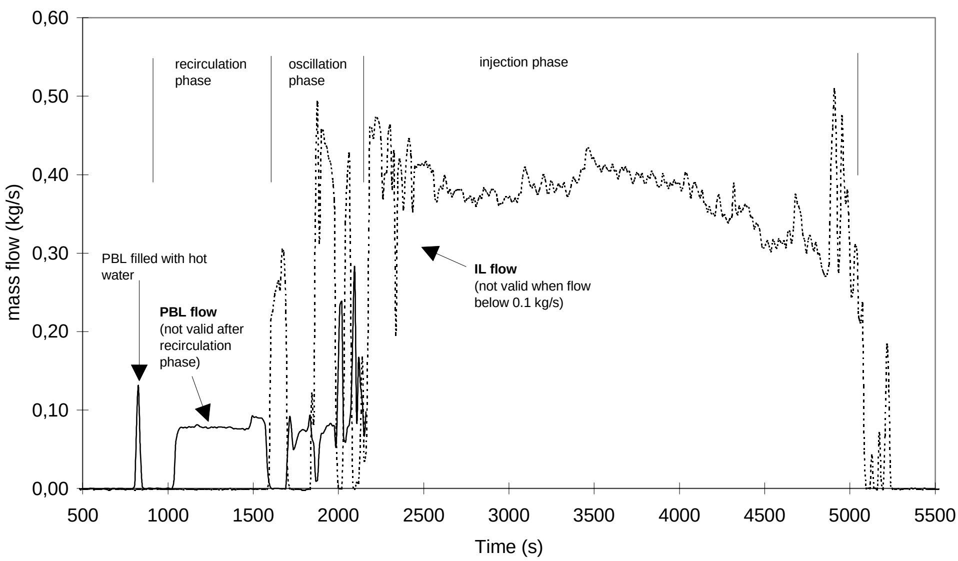

Figure 3. Mass flow rate through the PSIS.

Flow rate through the PSIS varied as the driving force changed during different PSIS operation modes. The flow was low during the recirculation phase and high during the injection phase. Between these two phases flow rate oscillated.

1.3.2 Parametric studies

The experiments studied the effects of **the break size and location, CMT size and position, removal of the flow distributor (sparger), the initial water temperature in the CMT and PBL, the PBL connection position and the injection line flow resistance** on the PSIS behaviour during SBLOCA's.

Break size:

During the recirculation phase, hot water flowed through the PBL to the CMT and replaced cold water there. The average recirculation flow rate decreased slightly by decreasing break size. The break size had large influence on the duration of the recirculation phase. The length of the recirculation phase increased as the break size decreased. This was important because the driving head for recirculation flow, the density difference between the PBL and IL, reduced when the hot water filled the CMT. In the experiment with the smallest break size, the whole CMT and the IL became full of hot water, the driving head for recirculation flow disappeared, and the flow through the PSIS stopped. In PACTEL, only one CMT was in operation. In the real plant, flow through the all CMT's would stop in similar conditions. In PACTEL, the fact the flow through the PSIS stopped did not effect the main function of the PSIS: the CMT began to inject water as the cold leg level dropped below the PBL connection.

Between the recirculation and injection phases the flow through the PSIS oscillated. The oscillations took place when the cold leg water-level was close to the PBL connection. During this phase, the level in the CMT dropped, but the PBL was only partially full of steam. In the experiments with smallest break size, the flow through the PSIS was in this oscillating region until the end of the experiments and the injection phase never really started. In generally, the injection phase began when the level near the PBL connection dropped so much that only steam could enter the PBL. The injection phase was shorter in the experiments with larger break size.

Break location

In LOCA’s, the flow through the broken cold leg may reverse. If the loop flow reverses in a AP600 type reactor in the loop which has the PBL connection, water below saturation temperature may flow from the downcomer through the broken cold leg to the PBL and, finally, back to the CMT. This may lead to condensation in the CMT. The PACTEL experiments demonstrated the flow reversal and the flow of cold water from the downcomer through the cold leg towards the PBL could occur. The temperature of water flowing through the cold leg was so high that no significant condensation in the CMT occurred. Flow reversal in the broken cold leg will affect only one CMT and the CMT’s connected to the intact loops of the reactor would remain unaffected.

When the break located in the hot leg or close to the steam generator outlet in the cold leg, water level stabilized near the PBL connection and the PSIS flow was in the oscillating mode for an extended time. CMT injection flow never reached the full value since the PBL did not become completely empty of water.

CMT size and position

The experiments used two CMT tanks with different size. The experiments also studied effects of the CMT position on the PSIS behaviour. In the third series the CMT was at 1 metre higher position than in the previous experiments. This increased the driving head for PSIS flow from about 6.6 to 7.6 metres. As expected, the core heatup occurred later in the experiments with the larger CMT. The time needed to empty the CMT was directly propotional to the CMT water volume. The elevated CMT position increased the recirculation phase flow. The flow through the PSIS was about 10% higher when the CMT was at the higher position. The flow increased since the PSIS pipeline pressure losses had to compensate for the increased driving head because of the higher CMT position. During the CMT injection phase, the flow through the PSIS was more unstable. The flow stagnated and oscillated, which made comparison of the flow rates between the experiments more difficult. Some increase of IL flow was obvious in the experiments with the CMT at higher position, but the increase was not as large as during the recirculation phase.

Flow distributor (sparger)

The purpose of the flow distributor (sparger) in the CMT was to diminish the possibility of rapid condensation in the CMT, such as was observed in the earlier passive safety injection experiments in PACTEL. In the current experiments, the sparger was in use in the CMT and there were no problems with rapid condensation. In some experiments, the IL flowrate stopped temporarily during the injection phase when steam condensation in the CMT reduced pressure there. Condensation occurred when water temporarily filled the cold leg near the PBL connection and flowed to the CMT. The injection flow resumed soon after the PBL connection position in the cold leg became free of water. The flow stagnations lasted typically no longer than 2-3 minutes.

The removal of sparger had a significant influence on the PSIS behaviour. In the experiment without the sparger, intensive condensation in the CMT occurred and the CMT pressure dropped when steam began to flow to the CMT and the PSIS injection phase begun. The rapid condensation in the CMT led to strong mixing of the hot and cold water layers in the tank, and the temperature profile in the CMT became uniform. The pressure in the CMT dropped close to athmospheric whereas the pressure in the primary loop remained unchanged.

Initial temperature in the PSIS

The AP600 reactor PSIS design includes initially hot water in the PBL and cold water in the CMT and IL. The AP600 PBL configuration is such that hot water from the primary circuit fills the line through natural convection without any heating equipments or operator actions. The PSIS may become full of hot water during plant normal operation if the IL check valve leaks. As an extreme case, the experiments also studied the situation where the PBL was initially full of cold water. The fact the CMT becomes full of hot water during normal operation of the plant eliminates the PSIS recirculation phase, since the driving force for the PSIS flow disappeares. In the experiment, the fact the PSIS was initially full of hot water eliminated the recirculation phase, but did not affect the injection phase. The injection phase begun as planned when the water level in the cold leg dropped below the PBL connection position. The PSIS worked as planned also in the experiment where the PBL was initially full of cold water. In this experiment there was no initial driving force for flow through the PSIS. The flow through the PSIS started, however, when the operators opened the IL check valve. Also, in this case, the PSIS fulfilled its main function of providing water to the primary circuit. The initial water temperature in the CMT affected the water level in the reactor pressure vessel and the timing of core heatup at the end of the experiments. Core heated up earlier in the experiments where the PSIS was initially full of hot water.

PBL connection position

In the Westinghouse AP600 reactor, the PBL connects the top of the CMT to the cold leg. In the Korean CP-1300 Reactor, the PBL connects the top of the pressurizer to the CMT top. The experiment programme included one test with the PBL conection to the pressurizer top. The CMT worked as planned also in this experiment. The were no problems with condensation in the CMT, although there was practically no water circulation phase through the PBL. The change of PBL connection position affected the primary coolant distribution and, hence, timing of the core heat-up at the end of the experiment. Core heat-up occurred earlier in the experiment with PBL connection to the top of pressurizer than in the similar experiment with the cold leg PBL connection. The reason was the accumulation of water in the pressurizer.

IL flow resistance

For one experiment, the PACTEL operators installed an extra flow orifice in the IL, to reduce the flow. The reduced flow from the CMT had an influence on the water-level behaviour in the core. Water-level in the core was lower when the orifice was in use. The core level settled to the hot leg loop seal bottom elevation, and remained there as long as there was water in the CMT. The higher flow resistance in the IL did not change the principle behaviour of the PSIS. All three PSIS operation modes were present, with lower flow rates only. During the injection phase, because of the higher flow resistance, the injection flow was not oscillating as much as earlier.

1.3.3 Heat transfer to the CMT wall

As the CMT recirculated, hot liquid flowed to the top and created a hot liquid layer. Since the flows in the CMT were small, there was very little mixing of this layer with the colder water and a thermally stratified hot liquid layer formed. Above this thermally stratified layer there was a layer of saturated liquid. The thickness of the layers depended on the break size. Since the CMT walls were initially cold, the hot liquid layer transferred heat to the walls. The measured wall heat flux profile depended on the thickness of the hot liquid layer. In the smallest break experiments, the hot liquid layer was thick and the maximum heat flux occured when the hot water layer passed the thermocouple position. In medium size break experiments, the heat flux showed two peaks. The first peak occured when the hot liquid layer passed the thermocouple. The second peak corresponded to the time when the top of the liquid layer passed the thermocouple and condensation began. In the large break experiment, the hot liquid layer was very thin and the two peaks coincided.

When the wall thermocouples were underwater, the heat transfer mechanism was natural convection from liquid to the wall. During this phase the heat transfer coefficient varied between 1 and 4 kW/m2-K. As the water surface passed the thermocouple, condensation began. The change in the heat transfer mode increased local heat transfer coefficient.

1.3.4 Thermal stratification in the CMT

During the recirculation phase hot water filled the upper part of the CMT. Since the walls of the CMT were initially cold, hot water transferred heat to the walls and a thermally stratifield layer formed. The temperature profile in the CMT was steep in the beginning of the transient but became smoother as the water level in the tank lowered. Condensation of steam and flow of water to the tank brought more hot water to the CMT and increased the hot liquid layer thickness. Flashing of hot liquid layer may also have happen when the pressure in the tank dropped due to steam condensation.

The thickness of thermally stratifield and saturated liquid layers varied in the experiments. In the large break experiments or when the break was at the outlet part of the cold leg, the saturated liquid layer was very thin and the water temperature in the CMT reached saturation close to the water surface. The thickness of saturated layer increased as the break size decreased or when the break position was changed to the inlet part of the cold leg or to the hot leg. See Figure 4 for an example of the positions of actual water level and the top and bottom of the thermally stratifield layer one experiment.

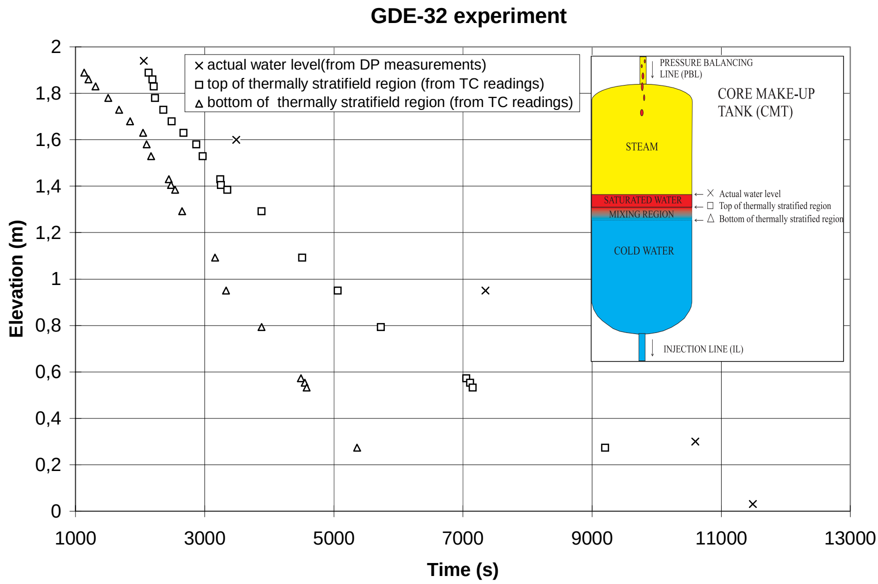

Figure 4. CMT water level in GDE-32. Position of thermally stratifield region detected using the temperature and differential pressure measurement data.

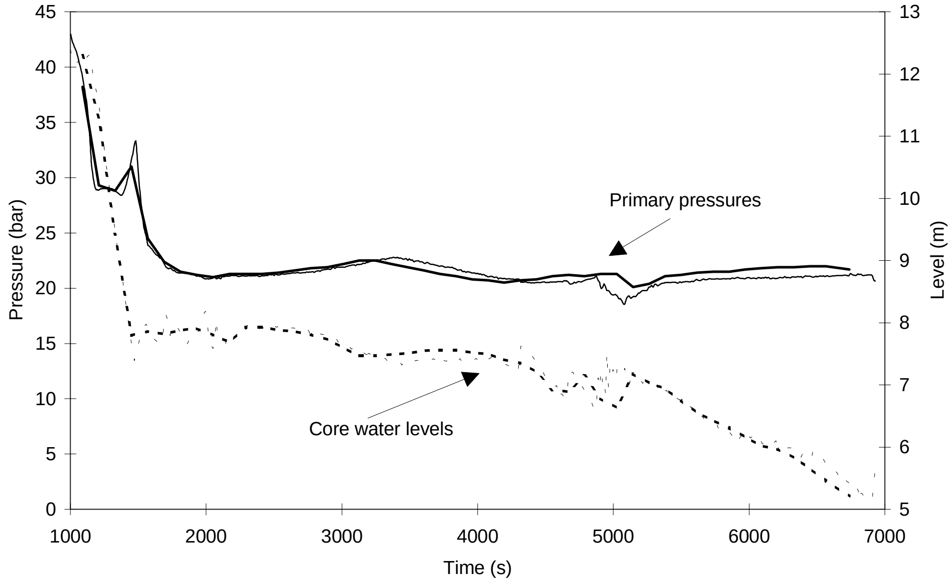

Figure 5. Primary pressure and core water level in the experiments studing reproducibility of the PSIS phenomena.

1.3.5 Reproducibility of the phenomena

To study reproducibility of the phenomena in the CMT, the experiment team repeated one experiment with identical parameters. The overall behaviour of the loop was similar in both experiments. The later phase of the experiments showed some differences in the primary and secondary pressures and in the pressuriser level behaviour. The pressure in the CMT dropped due to condensation in the both experiment when the injection phase started. The CMT draining times were about 100 seconds different. Consequently, the core heat-up was delayed by about 170 seconds. See Figure 5 for comparison of the primary pressure and core water level in the repeated experiments.

2. APROS SIMULATIONS OF THE EXPERIMENTS

PAHKO project included computer simulations of three selected experiments with APROS code. In addition, CATHARE and RELAP calculations were done as a part of the EC funded project. The details of the simulation results are available in the EC project reports. The following chapter highlights the results of the APROS calculations.

2.1 Results and Conclusions from APROS simulations

The APROS code was able to follow the simulated transients with good accuracy, provided that dense nodalization was used in the CMT. The code predicted the time for core heat-up within the measurement uncertanties i.e. within the difference observed between the two repeated experiments. The simulations included, however, some problems for detailed describtion of the CMT phenomena. The problems were similar to those observed in the CATHARE and RELAP calculations and partly followed from the numerical solution methods of the codes. Numerical diffusion led to smoothing of the temperature profile in the CMT and too high condensation rate in the tank. The calculated CMT injection rate oscillated due to condensation. The code was not capable for keeping continuously the uppermost water layer in the CMT in saturated conditions. The saturated conditions were reached only temporarely when the CMT water level passed a node boundary. Increasing the number of nodes partly eliminated this problem, but did not solve it completely. See Figures 6 - 8 for examples of APROS simulation results.

The heat losses became important in the simulations of long transients, such as the third simulation exercice (GDE-43 experiment). The distribution of heat losses between the pressurizer and upper plenum were important for the distribution of coolant in the primary loop. The fact that APROS calculated too much water in the pressurizer during the later phases of the transient was the reason for the fact the calculated recirculation flow did not stop.

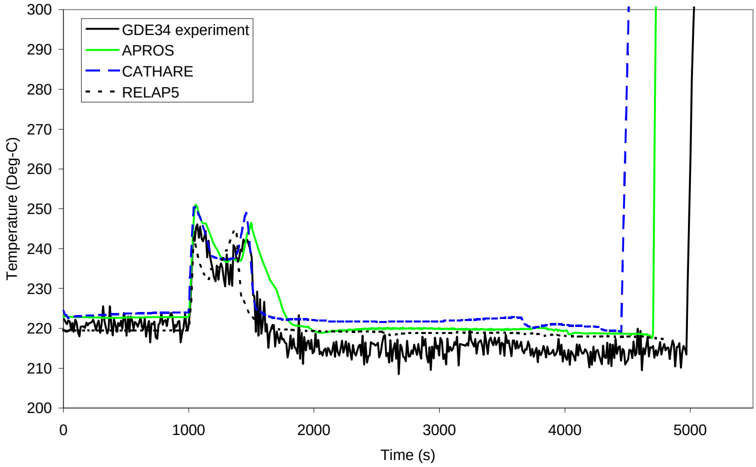

Figure 6. Comparison of the simulated and meassured rod cladding temperature at the core outlet in the GDE-34 experiment.

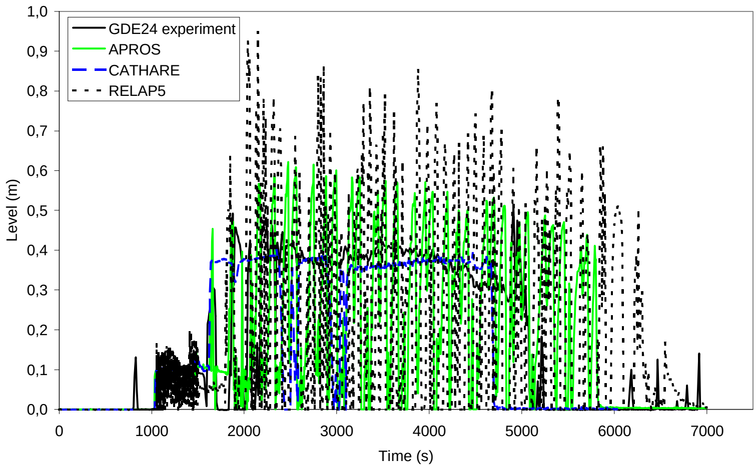

Figure 7. Comparison of the simulated and meassured CMT injection flow rates in the GDE-24 experiment.

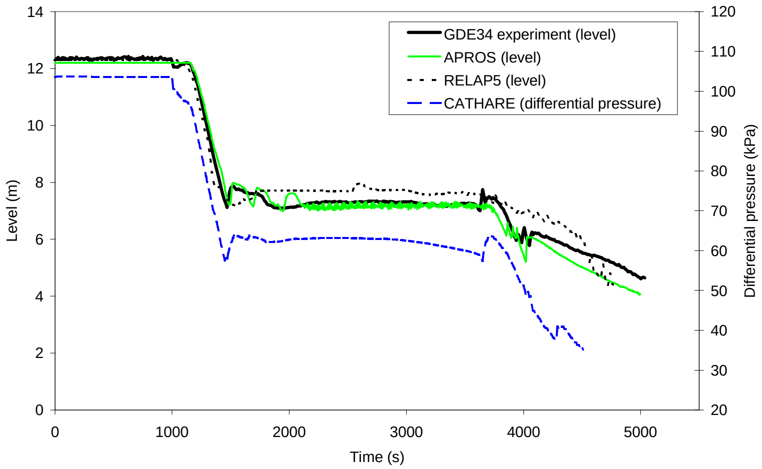

Figure 8. Comparison of the simulated and meassured core water level in the GDE-34 experiment.

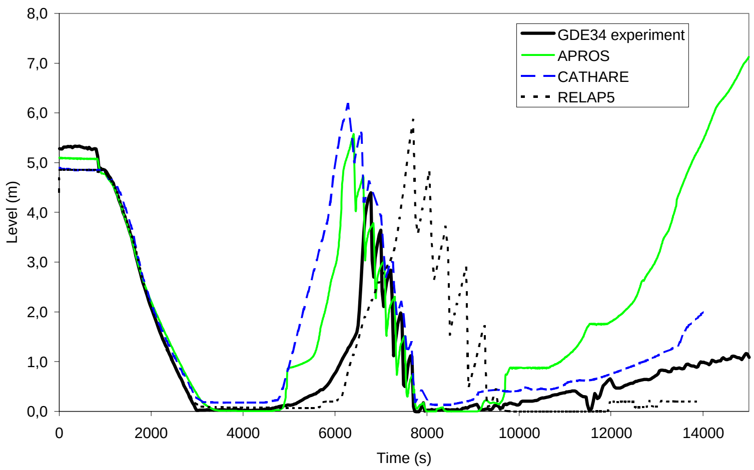

Figure 9. Comparison of the simulated and meassured pressurizer water level in the GDE-43 experiment.

3. CONCLUSIONS

The purpose of the PACTEL experiments was to investigate phenomena in the PSIS during SBLOCAs. The investigated PSIS worked as planned in all experiments provided the CMT was equipped with a flow distributer (sparger). The main source of disturbances for the investigated PSIS is condensation, which could occurs in the CMT when steam or two-phase mixture begins to flow to the tank, in the PSIS pipelines if countercurrent flow of steam and cold water occurs or near the ECC water injection position. In the current experiment series, condensation did not cause problems for PSIS behaviour as long as the sparger was used.

The computer simulations reproduced the measured transient behaviour with good accuracy in all three simulation cases. This was possible when the CMT was modelled with short nodes and, in the case of RELAP5 code, when the condensation rate in the CMT was artificially reduced. The simulations of the selected experiments included some problems in detailed calculations of thermal stratification and condensation in the CMT, and in the calculation of the PSIS flow when the driving force for the flow was small. Accurate simulation of these phenomen would require further improvements and testing of the code models.

Reference

Tuunanen, J., Vihavainen, J., D'Auria, F., Kimber, G., **Final Report of the European Commission 4th Framework Programme Project "Assessment of Passive Safety Injection Systems of ALWR's".** VTT Energy, Research Report. 15.11.1998.

Severe accident management: a summary of the VAHTI and ROIMA projects

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Abstract

Two severe accident research projects: “Severe Accident Management” (VAHTI), 1994-96 and “Reactor Accidents’ Phenomena and Simulation (ROIMA) 1997-98. have been conducted at VTT Energy within the RETU research programme. The main objective was to assist the severe accident management programmes of the Finnish nuclear power plants.

The projects had several subtopics. These included thermal hydraulic validation of the APROS code, studies of failure mode of the BWR pressure vessel, investigation of core melt progression within a BWR pressure vessel, containment phenomena, development of a computerised severe accident training tool, and aerosol behaviour experiments. The last topic is summarized by another paper in the seminar.

The projects have met the objectives set at the project commencement. Calculation tools have been developed and validated suitable for analyses of questions specific for the Finnish plants. Experimental fission product data have been produced that can be used to validate containment aerosol codes. The tools and results have been utilised in plant assessments.

One of the main achievements has been the computer code PASULA for analysis of interactions between core melt and pressure vessel. The code has been applied to pressure vessel penetration analysis. The results have shown the importance of the nozzle construction. Modelling possibilities have recently improved by addition of a creep and porous debris models.

Cooling of a degraded BWR core has been systematically studied as joint Nordic projects with a set of severe accident codes. Estimates for coolable conditions have been provided. Recriticality due to reflooding of a damaged core has been evaluated.

1. INTRODUCTION

The paper is a summary of the publicly funded nuclear reactor safety research projects “Severe Accident Management” (VAHTI), 1994-96 and “Reactor Accidents’ Phenomena and Simulation (ROIMA) 1997-98.

The main objective of the VAHTI and ROIMA projects was to assist the current Severe Accident Management (SAM) programmes of the Finnish nuclear power plants. The background for SAM were the comprehensive plant specific PSA level 1 studies available for both Finnish plants. The utilities operating the nuclear power plants had already implemented plant backfits and operating improvements to cope with severe accidents. Against this background, the objective of the VAHTI and ROIMA projects was set to investigate some selected questions in detail. 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 projects were divided into several main topics:

- thermal hydraulic validation of the APROS code,
- failure mode of the BWR pressure vessel,
- core melt progression within a BWR pressure vessel,
- containment phenomena,
- development of a computerised severe accident training tool, and
- fission product behaviour experiments.

A summary of the fission product experiments conducted within the VAHTI and ROIMA projects is given by another presentation in the seminar (Jokiniemi 1998). A summary report of the VAHTI research project has been published (Sairanen 1997).

2. THERMAL HYDRAULIC VALIDATION OF THE APROS CODE

Thermal hydraulic phenomena play a key role in the progression of transients and accidents into core damage. They are also essential when assessing the plant behaviour in the severe accidents. The thermal hydraulic computer codes are necessary tools in planning efficient SAM strategies and developing emergency operating procedures. Two thermal hydraulic codes have been used: the RELAP5 and APROS codes.

The Advanced Process Simulation Environment APROS (Puska et al. 1995) has been jointly developed by VTT and Imatran Voima Oy (IVO). The basic APROS development has been conducted in a separate programme financed by VTT and IVO. The task of the VAHTI and ROIMA projects has been thermal hydraulic testing of the APROS models. The APROS based nuclear power plant analysers of the Loviisa and Olkiluoto plants have also been improved.

2.1 Testing of interfacial friction

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) has been calculated with several code versions and modelling approaches. The experiment was chosen as a high priority case, because correct modelling of a loop seal is essential for simulating small break accidents in the VVER-440 reactors. The results and observations were utilised in development of later APROS versions (Karppinen 1996a).

The LOTUS air/water annular flow experiments (Owen et al. 1985) were another source for interfacial friction model testing. Two pressure loss and water entrainment tests were calculated. The water entrainment was in good agreement with the experimental data. The two phase pressure drop was well predicted with low air fluxes but systematically overpredicted with higher air fluxes. (Karppinen 1996b).

2.2 The Olkiluoto BWR model

A transient analysis model of the Olkiluoto BWR has been created with APROS (Puska et al. 1997). The nodalization of the reactor vessel is shown in Fig. 1.

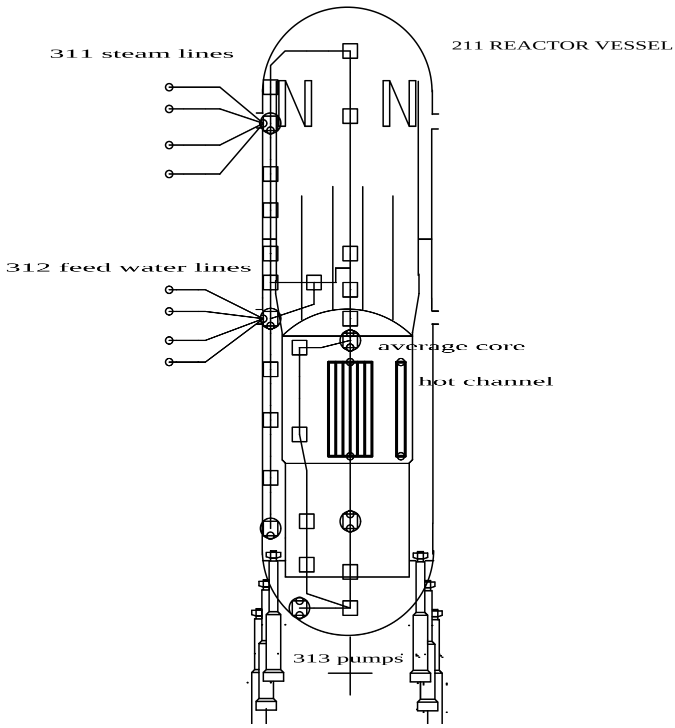

Figure 1. Reactor vessel nodalization in the APROS Olkiluoto BWR model.

Three transients have been analysed with the Olkiluoto model:

1. double ended steam line break with simultaneous loss of electricity
2. loss of feed water, ATWS
3. simultaneous closure of all steam line isolation valves

The results showed that this type of BWR transients can be calculated with APROS using the one-dimensional core model in combination with the available thermal-hydraulic models. The analyses also indicated improvement needs for further BWR studies:

- the wall heat transfer, interfacial mass transfer and boiling incipience models should be improved,
- the phase separation models should be validated,
- wet steam thermodynamics should be improved, and
- a mechanistic steam separator model should be developed.

3. THE REACTOR PRESSURE VESSEL FAILURE MODE

The PASULA code system has been developed for the pressure vessel lower head integrity analyses in severe accidents. A special incentive has been investigation of the pressure vessel failure mode. The question is important due to the SAM adopted for the Olkiluoto BWRs, which specifies flooding of the lower drywell prior to the vessel failure. The bottom head failure mode is one of the key parameters to determine the extent of melt-water interaction.

The PASULA system (Ikonen 1997) consists of a set of codes for specific tasks. Heat conduction and convection analysis can be performed for two- and three-dimensions. 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 analysed.

Analysis of the solid continuum is performed by 2D and 3D finite element codes capable for calculation of elastic and plastic deformations. Material creep and large deformations can be analysed. An arbitrary geometry can be modelled with the three-dimensional system. The codes are also suited for fracture mechanical analysis, such as calculating the J-integral surface cracks in the reactor pressure vessel. Analysis of shallow cracks can also be performed.

The PASULA code has been applied to calculate pressure vessel mechanical behaviour tests (CORVIS, Rupther). It has also been utilised in detailed studies of the Olkiluoto plant pressure vessel bottom head penetrations in core melt accidents. The objective was to compare different penetration types and to evaluate their potential for initiating a vessel failure. The calculations showed that the construction of a nozzle is central on the failure mode. Main factors are location of load carrying points, wall thickness and the materials involved.

A general conclusion was that the results were not sensitive to the amount of corium on the vessel bottom head. A comparison of the instrument tube and control rod penetration calculations showed that a large corium leakage through the control rod penetrations is not expected for this type of BWR pressure vessel. On the contrary, walls of an instrument nozzle tube will fail quickly, within some tens of seconds. The instrument tubes are not jointly supported, and ejection of one of them is much more probable than that of a control rod tube. The vessel failure is hence expected to be initiated at the instrument tubes.

The studies assumed that the core debris in the lower head was liquid. This is plausible if a large quantity of melt discharges into the lower head. An alternative scenario is melt draining in small batches resulting to debris quenching and formation of a rubble bed. A heat conductivity formulation for granular media was developed to analyse the latter scenario. The new model has been described in detail by Ikonen & Lindholm (1996).

The instrument tubes were further investigated by Lindholm et al. (1998) utilising the porous debris model. The melt migration into the lower head was calculated with the MELCOR and MAAP4 codes and the resulting boundary conditions were input for the PASULA code. The core debris was calculated to cool down during migration through the water pool. Response of the instrument and control rod tubes in different locations (Figure 2) were then studied. The PASULA calculations indicated that the lower head failure would most probably initiate at the outer ring (2,4) location. The failure location is determined by the debris composition, which is more oxidic in outer rim. The failure would be initiated in an instrument tube about 1 h after the lower head dryout.

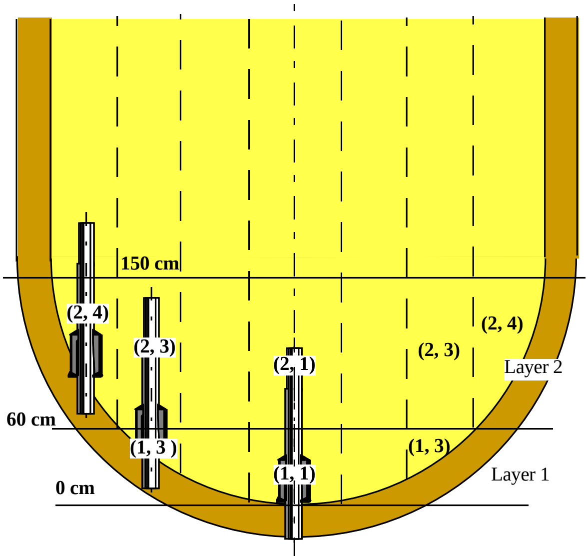

Figure 2. Location of the penetrations analysed with the PASULA code (Lindholm et al. 1998).

A recent addition to the PASULA system is modelling of material creep. A detailed description of the model has been published by Ikonen (1996). The code has been first applied to calculate creep of Zircaloy fuel rod claddings under fast transients (Ikonen et al. 1997). Validation of the creep analysis method for reactor pressure vessel type of stainless steels is under way. The three-dimensional Rupther test conducted in the EC Revisa project has been calculated. The ultimate goal is to predict the failure mode of a reactor pressure vessel under all important severe accident scenarios.

4. BWR CORE DEGRADATION

Consequences of operator actions after a partial core damage have been systematically evaluated in a series of studies conducted as a part of the RAK-2 project within Nordic Nuclear Safety Programme (NKS). The objective was to investigate when and how the reactor core is still coolable if safety systems become available during the accident, and what are the consequences. The reflooding calculations (Lindholm et al. 1995a) indicated that a short time window exists between relocation of the BWR control materials and fuel degradation. During this interval, the core would be in a reactive configuration without absorbing materials. The emergency core cooling water in the Nordic BWRs does not contain boron, which raised the question of BWR recriticality. The third step is to investigate if the core debris reaches 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.

4.1 Reflooding of an Overheated Core

Reflooding of an overheated BWR core was investigated by calculating accident scenarios for ABB Atom BWRs with three different computer codes: MAAP4, MELCOR 1.8.3, and SCDAP/RELAP5 (Lindholm et al. 1995a, and Lindholm et al. 1995b). The calculations were carried out for the Finnish Olkiluoto and Swedish Forsmark plants, which resemble each other and use same type of fuel and control rods, but that have different thermal powers and different alignments of safety injection systems.

The base scenario was a station blackout with two variations:

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

A variety of times was assumed for restoration of power and subsequent start of coolant injection. The important phenomena include rapidly increased hydrogen generation and the recriticality considerations.

Rapid core cooling was obtained in cases when reflooding was started at maximum core temperature of < 1600 K, even with only 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. All the codes predicted a different end state of core after reflooding: MAAP4 predicted formation of melt pool in the core, MELCOR resulted in formation of rubble bed and SCDAP/RELAP5 predicted material melting or fuel fragmentation due to mechanical stresses.

The emergency coolant injection caused hydrogen production peak in all reflooding cases. The total estimated hydrogen masses with different assumptions are presented in Table 1.

Table 1. Hydrogen production after reflooding of a degraded core. Olkiluoto station blackout (Lindholm et al. 1995a).

Case	Total hydrogen production during the transient	
Low pressure cases	MELCOR 1.8.3 results	MAAP 4 results
Base case. no reflood	424 kg [1)]	110 kg
Reflood at 1400 K	373 kg	240 kg
Reflood at 1600 K	445 kg	450 kg
Reflood at 1800 K	413 kg	410 kg
Reflood at 2000 K	500 kg[1)]	490 kg
High pressure cases (No ADS)	MELCOR 1.8.3 results	MAAP 4 results
Base case. no reflood	730 kg[1)]	500 kg
Reflood at 1400 K	32 kg	63 kg
Reflood at 1600 K	90 kg	150 kg
Reflood at 1800 K	90 kg	140 kg
Reflood at 2000 K	90 kg	140 kg

[1)] value at core support plate failure

All codes predicted some time window between melting of control rods and fuel. The length of the time window varied with codes, being 1-2 min with MELCOR and 3 -15 min with MAAP4. During this period the fuel pellets may be in a rubble bed or in intact fuel geometry. The subsequent recriticality studies were judged to be most interesting in cases, where fuel rods are in intact geometry and the control rods have partially melted.

4.2 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 RPV lower head water pool. The BWR lower head houses a number of penetrations in the lower head bottom. These 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.

A number of calculations to study the modelling parameters was conducted with the MELCOR code (Lindholm 1996). Both low and high pressure scenarios were investigated. In general, reflooding after lower head dryout accelerated the lower head failure. This was caused by excessive metal oxidation in the debris bed, which overrides the cooling effect of water. The interval between lower head dryout and failure of the instrument tubes varied from 13 min to 70 min with different debris particle sizes and initial porosity assumed. The results may be specific of the MELCOR modelling. If the MELCOR model is correct, reflooding after lower head dryout will not prevent instrument tube failure.

4.3 Recriticality in a Degraded Core

Joint Nordic recriticality studies have been performed (Højerup et al. 1997, Miettinen et al. 1998). Severe accident codes were used to give estimates for control rod/fuel configurations at the start of reflooding and reactor physics codes were applied to calculate reactivity effects during the transient. Three codes: RECRIT, APROS and SIMULATE-3K were used for tasks requiring reactor physics. The RECRIT code was initially written at Risø National Laboratory as a neutronics code. Thermal hydraulic models developed at the VAHTI project were later combined into it to form the present RECRIT code version.

The application case was a BWR accident initiated with total loss of electricity. Several coolant injection rates into the degraded core 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 coolant is entrained above the quenching front. Criticality may be expected when the quenching front reaches a core zone without control materials.

The results showed that reflooding of a partly control rod free core gives a recriticality power peak of a high amplitude but 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 again and tend to stabilise on a level that can be some tens per cent of normal power.

5. CONTAINMENT PHENOMENA

5.1 Melt Coolability

Melt coolability in the containment is one of the key questions in severe accident management. The Finnish utilities and VTT Energy have participated in the MACE research programme, in which melt coolability has been experimentally studied by using real reactor materials. Six MACE experiments have been performed at Argonne National Laboratory since 1989.

Applicability of the MACE programme results to Finnish plants has been evaluated by Lindholm (1998). The evaluation considers the general observations seen in the MACE tests, scaling aspects, and material differences between the MACE experiments and Finnish plants. Heat transfer characteristics have been derived from the early MACE tests M0 and M1b. The results have been used to produce coolability estimates for the plant specific accident sequences in the Finnish plants. Typically the core melt would cool from the 2500 K initial temperature to the solidus temperature of the silicate concrete, 1400 K, in 2-3 hours both in case of the Olkiluoto and Loviisa plants. The concrete floor would be eroded in this time by about 10 cm.

5.2 Hydrogen Mixing

The CONTAIN code has been tested against the natural circulation and helium mixing experiments performed at the VICTORIA containment test facility. The VICTORIA facility (Lundström et al. 1996) has been built by IVO to specifically study accident management aspects of the Loviisa ice condenser containment. One of the hydrogen management actions is to open the ice condenser doors to promote natural circulation and hydrogen mixing. The concept was studied by VICTORIA experiments. A global natural circulation loop through the ice condensers has been observed.

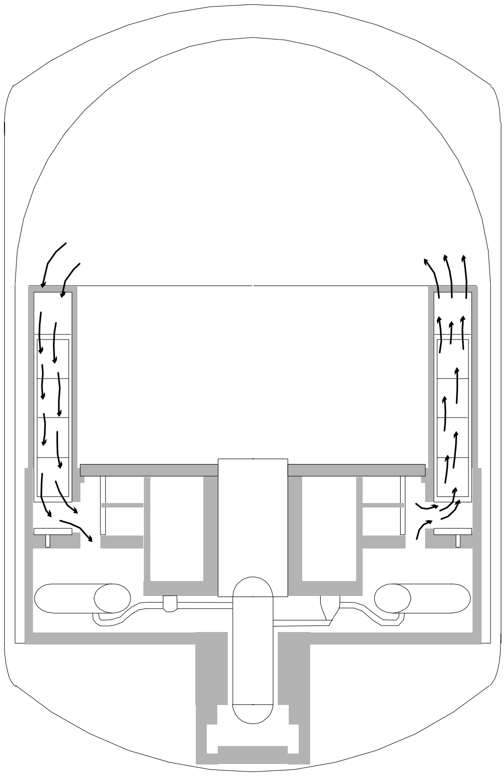

Figure 3. The global circulation flow observed in the VICTORIA experiments.

A global natural circulation flow as in Fig. 3 could be predicted by the CONTAIN code relatively well. The ice depletion time was also in good agreement with the experiment. The code slightly overpredicted the containment pressure and temperatures, especially in the late phase of the experiments. This is presumably due to energy balance mismatch, but the exact source of the discrepancy could not be solved.

5.3 Direct Containment Heating

Containment thermal hydraulic loads during high pressure melt ejection has been parametrically studied for the ABB Atom BWRs (Silde & Lindholm 1997). The calculations have been conducted by using the CONTAIN and MELCOR codes within the joint Nordic NKS/RAK-2 research project. Metallic and oxidic debris sources have been considered.

The highest calculated pressure peaks were about 8.5 bar both in case of metallic and oxidic debris ejection.

Very high gas temperatures, about 1900 K, were calculated in the pedestal atmosphere in case of metallic debris. The high temperatures resulted from oxidation of the metallic particles by superheated steam released simultaneously with debris. However, estimates for the metallic debris were sensitive for the model parameters, like particle size and chemical reaction kinetics.

Gas temperatures in the containment were considerably lower, 400-500 K, in case of an oxidic debris ejection,. The estimates were quite insensitive to parametric variations. The containment gas temperatures were mainly determined by the large amount of steam that was released into the containment as a consequence of the melt ejection.

6. ACCIDENT MANAGEMENT SUPPORT

Plant modifications for severe accidents and new SAM operating procedures are to be completed for Finnish plants within a few years. A simulation tool for training of operators and technical support personnel is needed when the new systems are introduced. The training tool will be developed by implementing selected severe accident models into the APROS environment. The focus in the first stage is to model the severe accident management actions for the Loviisa plant.

Models have been developed in the VAHTI and ROIMA projects. The work was begun by improving the APROS containment model to describe the physical conditions and accident management hardware relevant for severe accidents. At the same time, all containment submodels of the older code version were reviewed. Some important severe accident features are still lacking. The most notable ones are models for hydrogen recombiners and gas combustion. Work is in progress to add these models into the APROS system.

Status of the APROS containment model has been described in (Silde 1998). Main submodels and the new features under development or considered are shown in Table 2.

Table 2. Characteristics of the APROS containment model.

MODEL	YES	NO	DEVELOPMENT PLANNED
Steam/non-condensable gas mixture thermodynamics	x		
Water droplet host (fog)	x		
Intercell gas flow	x		
Buoyancy effect	x		
Gas diffusion		x	x
Intercell water flow	x		
Intercell flow of fog droplets		x	
Heat and mass transfer between gas region and heat structures, between gas region and water droplets, between gas region and water pool	 x x x		
Condensate film tracking		x	x
Heat transfer between water pool and heat structures	x		
Pool boiling heat transfer		x	
Pool scrubbing		x	
1-D and 2-D transient conduction within 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	x
Fission product behaviour		x	x
Cavity phenomena (MCCI)		x	x
Fan coolers		x	
Gas combustion		x	x
Direct containment heating		x	
Hydrogen igniters		x	x
Hydrogen recombiners		x	x
Thermal radiation		x	x

7. SUMMARY

The "Severe Accident Management" (VAHTI) 1994-96 and "Reactor Accidents' Phenomena and Simulation" (ROIMA) 1997-98 projects have been conducted within the RETU research programme. The main objective of the projects has been to assist the severe accident management programmes of the Finnish nuclear power plants by investigating selected questions in detail. The projects were divided into several subtopics:

- thermal hydraulic validation of the APROS code,
- failure mode of the BWR pressure vessel,
- core melt progression within a BWR pressure vessel,
- containment phenomena,
- development of a computerised severe accident training tool, and
- aerosol behaviour experiments

The projects have met the objectives set at the project commencement. Calculation tools have been developed and validated suitable for analyses of questions specific for the Finnish plants. The tools and results have been utilised in plant assessments. Experimental fission product data have been produced that can be used to validate containment aerosol codes.

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FISSION PRODUCT BEHAVIOUR IN SEVERE ACCIDENTS

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1. Introduction

The understanding of fission product (FP) behaviour in severe accidents is important for source term assessment and accident mitigation measures. For example in accident management the operator needs to know the effect of different actions on the behaviour and release of fission products.

At VTT fission product behaviour have been studied in different national and international projects. In this paper we report the results of projects in EU funded 4th framework programme Nuclear Fission Safety 1994-1998. The projects are: fission product vapour/aerosol chemistry in the primary circuit (FI4SCT960020), aerosol physics in containment (FI4SCT950016), revaporization of test samples from Phebus fission products (FI4SCT960019) and assessment of models for fission product revaporisation (FI4SCT960044). Also results from the national project “aerosol experiments in the Victoria facility” funded by IVO PE and VTT Energy are reported.

Our results on aerosol behaviour in containment (AHMED and VICTORIA facilities) together with results from other facilities complete our understanding on soluble and non-soluble aerosol behaviour at known thermal hydraulic conditions. The existing uncertainties lie on predicting the flow patterns, temperature fields, water vapour concentration and composition of aerosol source into the containment. There are larger uncertainties in fission product behaviour in reactor coolant system (RCS). The chemical forms of fission products transported through the RCS are important for iodine partition between gaseous and aerosol phase, retention in the RCS, late phase revaporisation of FPs and aerosol behaviour in containment. In the work presented here we have gained more knowledge on these phenomena, but large uncertainties still exist in FP release and behaviour at high temperatures in RCS.

2. Primary Circuit Chemistry

The behaviour of Cs vapours (CsOH and CsI) was studied under conditions relevant to primary circuit during a nuclear reactor accident. Experimental work was carried out using laminar flow reactors with a diameter of 8 cm and a length of 1.6 - 2 m. This work consisted of two separate research efforts. The chemical reactions of CsOH and CsI vapours with metal aerosol particles (simulating those formed of structural and control rod materials during an accident) were studied in temperature up to 1600 °C. In the other part of the project, the aerosol particle formation from CsOH vapour was studied at a temperature of below 600 °C (Valmari et al., 1998).

2.1 Chemical reactions of CsOH and CsI vapours with metal particles and vapours

Aerosol particles were generated from the water-solutions by ultrasonic nebulizers and transported into the laminar flow reactor in a carrier gas consisting of 93 or 99 mol-% N_2 and 1 or 7 mol-% H_2. During each test, only two reactants were present (CsOH or CsI + one of the metals Ni, Cr, Fe or Ag). As the gas was heated, water was evaporated from the particles, CsI / CsOH was evaporated and the metal nitrate was decomposed. According to thermodynamic equilibrium calculations, Ni and Ag formed elemental compound and Fe and Cr oxidic compounds.

2.2 Particle size distributions (ELPI)

Aerosol particle concentrations were measured continuously with a prototype version of an Electrical Low Pressure Impactor (ELPI, Keskinen et al., 1992). The particle number concentration was 2 - 3 $\times$ 10^8 particles/cm^3 when a volatile compound (CsOH or CsI) was fed into the reactor together with the metal particles, that is of order of 100 higher as compared to the concentration of the original metal particles (Fig. 1). This indicates that CsOH and CsI were not completely reacted with or condensed on the metal particles but formed new particles via homogenous nucleation.

2.3 Elemental composition

The water-soluble and insoluble fractions were analysed separately from the aerosol particle samples collected on Nuclepore filters (Table 1). Iron did not react with CsI to a significant extent, as 99.9 % of Fe was present in water-insoluble form, whereas 98 - 99 % of Cs and I were water-soluble. Iron could have been present as Fe, FeO or Fe_3O4, all of which are practically insoluble in water. Cs and I were presumably present as CsI. The molar ratio Cs / I was 1.10, suggesting that CsI was not decomposed in significant extent to CsOH and gaseous iodine.

Ni reacted with both CsOH and CsI. 3 - 6 % of Ni was found in water-soluble form. The reaction product including both Cs and Ni could not be deduced using thermodynamic models, because thermodynamic data was not available for any

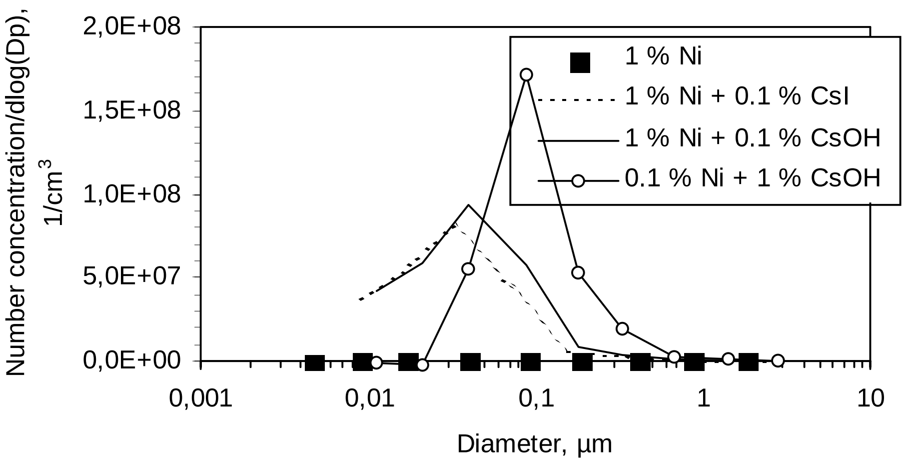

Figure 1. Size distributions produced by different vapours (with quenching). Percentage in the legend refers to concentration in the water-solution in the nebulizer.

such compounds. In the case of Ni + CsI experiments, the reaction product may have been NiI_2, although it is not preferred according to thermodynamic assessments. During experiments with CsI, iodine was also almost entirely found in water-soluble form and the molar ratio Cs / I was close to unity.

Table 1. Elemental analysis results of the filter samples. Analyses were carried out with ICP-MS method.

Reactants	T, °C	Water-soluble fraction, %					Molar ratio
		Fe	Ni	Cr	Cs	I	Cs / I
CsI + Fe	1300	0.07	-	-	98	99	1.10
CsI + Ni	1300	-	6	-	98	98	1.07
CsOH + Ni	1300	-	3	-	97	-	-
CsOH + Cr	1300	-	-	[a]	98	-	-
CsI + Cr	1300	-	-	[a]	98	97	1.19
CsOH + Cr	1000	-	-	[a]	98	-	-
CsI + Cr	1000	-	-	[a]	98	91	3.13
CsI + Cr [b]	1000	-	-	[a]	98	98	1.14

a) Approximately 99 % of Cr was not observed neither in water-soluble or in insoluble fraction.

b) The amount of Cr fed into the reactor was about 10 % of the amount in the other samples.

The observed amount of Cr was only of the order of 1 % of the amount of Cr fed to the reactor, indicating that Cr was not soluble in water nor in the mixture of nitric acid and hydrofluoric acid, where the non-water-soluble fraction was dissolved. The absence of water-soluble Cr excludes the formation of water-soluble cesium chromates, such as Cs_2CrO_4. Chromium was probably present as Cr_2O_3. One of the CsI + Cr tests at 1000 °C showed a molar ratio Cs / I of 3.13 indicating that 2/3 of I was not present neither in the water-soluble or in the insoluble fraction. The un-observed I may have been associated with Cr in a compound not soluble in nitric acid or hydrofluoric acid.

2.4 CsOH aerosol formation and growth in laminar flow

CsOH was placed in a stainless steel vessel located inside the laminar flow reactor. The temperature was about 580 °C at the CsOH vessel outlet. The number size distribution of the particles formed by the nucleation of CsOH while the temperature was decreased was measured with a system consisting of a DMA and an UCPC (TSI model 3027) downstream of the reactor.

The particles formed a bimodal size distribution. The mean diameter of the major mode was 70 nm. The particle number concentration in this mode was 9 * 10^7 particles / cm^3 and the mass concentration 500 mg/m^3. In addition, an ultrafine particle mode at $d < 70$ nm was found to be formed via nucleation while the aerosol was quenched downstream the reactor. However, the ultrafine mode contained less than 1 % of the total mass in all the cases studied. No significant difference in the mass size distribution was observed when the temperature gradient was varied in the range of 5 - 8 °C/sec.

The particle number concentration was 7 - 8 $\times$ 10^7 particles / cm^3 in the Ag + CsOH experiments but only 1.0 $\times$ 10^7 particles / cm^3 when Ag was fed without CsOH (Fig. 2). This indicates that new particles were formed via nucleation of CsOH, despite the presence of Ag particles. The concentration of particles coarser than 130 nm was higher in the Ag + CsOH test as compared to the test with CsOH alone. This is likely due to CsOH vapour condensation onto the Ag seed particles instead of reactor walls in the case of higher concentration. The total mass concentration downstream the reactor was 545 mg/m^3 without seeds and 695 mg/m^3 with seeds, when the Ag contribution (195 mg/m^3) is excluded. Thus, the presence of seed particles increased the mass of CsOH not deposited on the reactor walls by 28 %. The formation of ultrafine particles via nucleation during quenching caused the concentration of particles with $d < 130$ nm to be higher without Ag particles. The mass concentration of particles in this size range was 52 mg/m^3 with and 90 mg/m^3 without seeds.

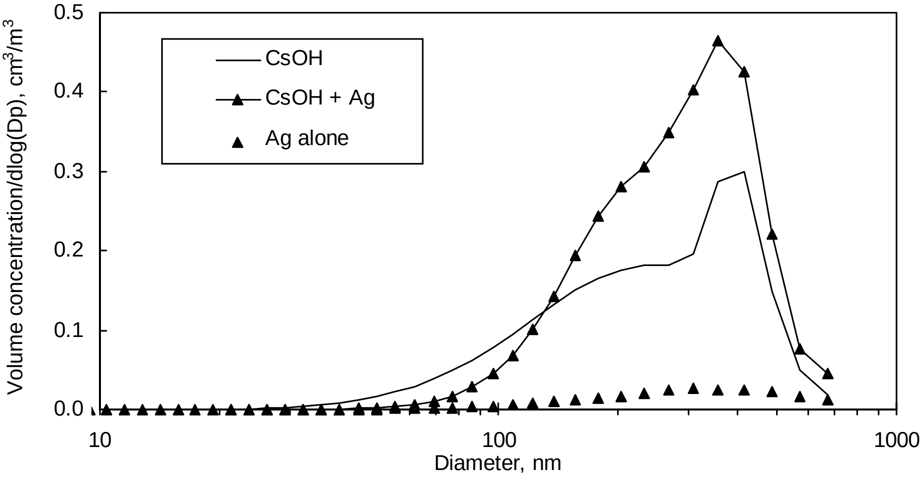

Figure 2. CsOH particle volumetric size distributions with and without Ag seed particles.

Commercial CFD-code FLUENT (Fluent Inc., 1996) with aerosol dynamics coupled into the code was utilised when interpreting the experimental results. The particle number concentration obtained by model calculation was underestimated by a factor of 10^4. The need for enhancement of the classical nucleation rates has also been reported by other researchers (e.g. Nguyen et al., 1987). The modelling of CsOH vapour behaviour was found to contain uncertainties due to inadequate data for the surface tension of CsOH. Also, the thermodynamic data given by different sources for CsOH varies causing remarkable uncertainty in the CsOH vapour pressure (Hontañón et al., 1996).

3. Radio-Tracer Studies on Fission Product Revaporisation in Severe Accident Conditions

To study the revaporization phenomena, two thermal-gradient tube facilities have been constructed for fission product vaporization rate measurements: one for use with synthetic radiolabelled samples at VTT and the other for analyses of the active Phebus-FPT-1 (Krischer and Rubinstein, 1992) samples at the CEC Joint Research Centre, Karlsruhe. Particular care has been taken to produce test conditions close to those expected in a severe accident. In the experiments, CsOH with a small amount of radioactive tracer is vaporized in a pure steam atmosphere from a sample plate at carefully controlled conditions. The main aim of these studies is to obtain the revaporization rate of CsOH as a function of temperature by using a gamma detector. To find the effect of the complex fluid dynamics of the system, it is modelled with the FLUENT CFD-software. Thus the complete velocity, temperature and vapour concentration profiles are obtained. Combining the experimental results with a detailed fluid dynamics simulation allows us to obtain simple engineering correlations, that are especially useful for severe accident computer codes like RAFT, VICTORIA and MELCOR (Auvinen et al., 1998).

3.1 Experimental

3.1.1 Facility description

A thermal-gradient tube (TGT) facility has been constructed at VTT to measure vaporization rates of synthetic fission product samples. With the TGT set-up stable thermal-hydraulic conditions can be achieved for the rate measurements. By applying a special porous tube dilutor vaporised radionuclides can be trapped in an aerosol filter and contamination of the facility is avoided. The experimental facility is presented in figure 3.

In an experiment the sample is put on a pre-oxidised AISI-304 stainless steel sample plate, which is placed in the TGT over the second heating element, 100 mm from the outlet of the furnace. A thermocouple controlling the temperature of the furnace is in contact with the sample plate. In addition, the temperatures of both heating elements are measured. During the experiment the temperature of the furnace is increased linearly from 20°C to 1000°C. The vaporisation rate, which coincides with the decreasing activity of the radio-labelled sample, is measured outside the furnace with a germanium gamma detector. Each integration lasts for about 70 seconds thus defining the time resolution. Since no standard is applied in the experiment, the first measurement is used as the calibration of the equipment.

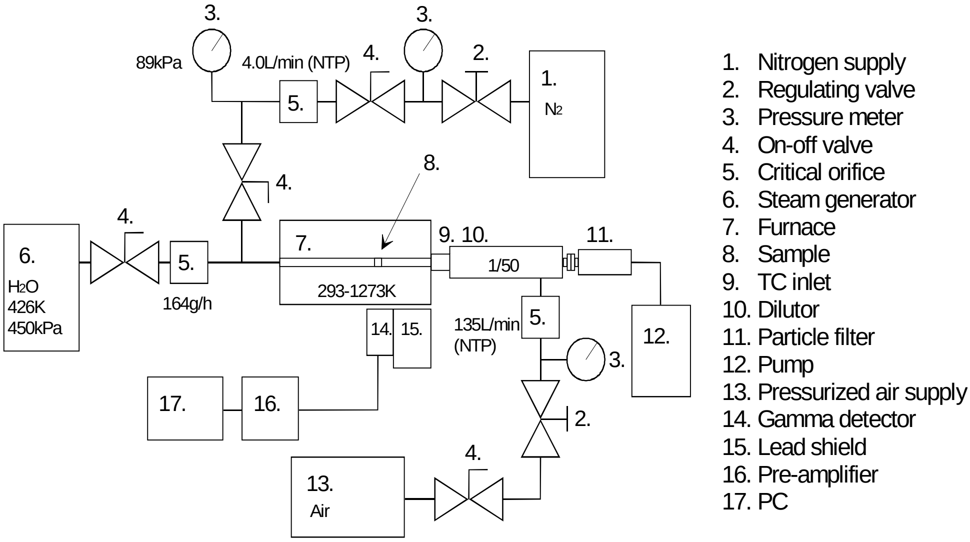

Figure 3. The experimental facility to measure fission product revaporization rates

Samples are vaporised in a pure steam atmosphere at 89 kPa pressure. The mass flow rate of the steam is fixed to 4.6E-5 kgs^{-1}. The flow velocity inside the furnace is therefore dependent on temperature and at 500°C it is approximately 0.6 ms^{-1}. Downstream of the furnace the gas flow is cooled with ambient air in a porous tube dilutor. The dilution ratio of the mass flow rate is close to 1/50. During the cooling process fission product vapours form aerosol particles, which are collected on a particle filter.

The day following the experiment the apparatus is disassembled. After removing each part, the remaining activity is measured for 30 min to resolve the effect of deposition on the test results. To quantify the deposition in the apparatus all components are measured also individually after dismantling.

3.1.2 Results

Two successful experiments to study the vaporisation of CsOH have been conducted so far. The geometry of the furnace and the sample plate were identical in those experiments. The temperature of the sample was raised up to

1000°C and the steam flow remained the same. However there were also a number of important differences.

The mass of the sample was increased from 25 mg in the first experiment to 803.5 mg in the second experiment. The radio tracer was changed from Cs_{134} containing $CsNO_2$ to Cs_{137} containing CsCl. This increased the initial activity from 2100 counts min^{-1} in the first experiment to 17900 counts min^{-1} in the second experiment and thus improved the statistical accuracy. Steam injection into the furnace was initiated at 500°C in the first experiment and at 300°C in the second experiment. The temperature ramp was decreased from 10°C min^{-1} to 2°C min^{-1} in the second experiment improving the temperature resolutions from 11.7°C to 2.3°C, respectively. In addition, the thermocouple inlet between the furnace and the dilutor was heated up to 482°C in the second experiment to decrease condensation of CsOH to that part. As a result of these changes the second experiment resulted in more accurate data than the first especially in respect of temperature resolution. The reduction of the sample mass from the second experiment is presented as a function of temperature in figure 4. Experimental data is also compared with the modelling results in the figure 4.

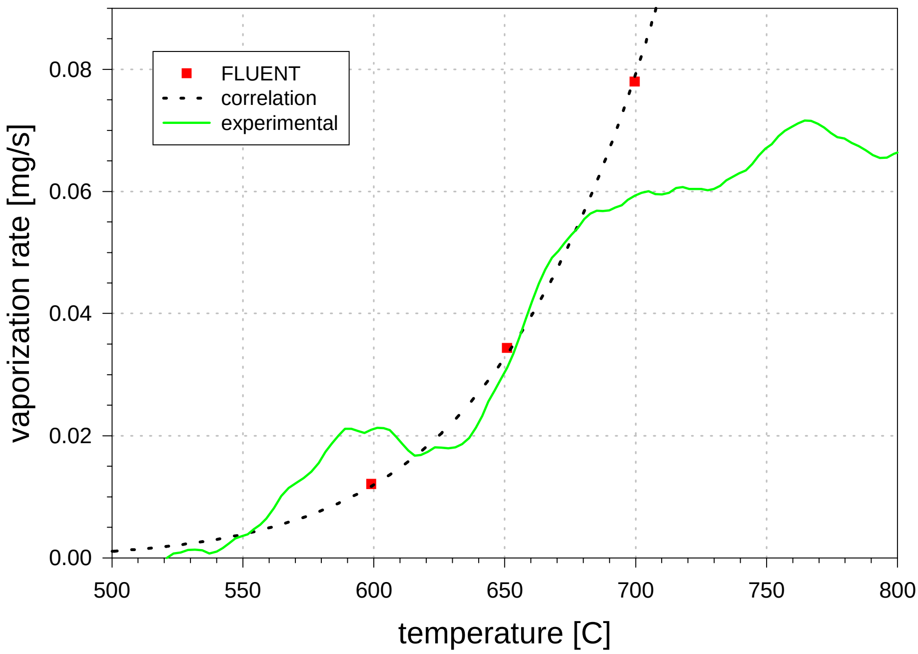

Figure 4. *Experiment 2: CsOH vaporization rate as a function of temperature*

3.1.3 Conclusions

A thermal-gradient tube (TGT) facility was constructed for fission product vaporization rate measurements to use with synthetic radiolabelled samples. The test facility has proven successful in measuring the temperature dependence of revaporization of CsOH: Vaporization starts at approx. 500°C and increases with temperature.

The flow, temperature and concentration fields of the system were simulated numerically with the FLUENT CFD-software. The experimental results and FLUENT modelling agree well in the temperature range 500°C - 700°C. Above this temperature, CsOH starts to react with the processing tube and the detected signal is affected by the deposition. To overcome this problem, further experiments with radiolabelled samples will be conducted with an improved gamma detector set-up.

A simple engineering calculation agrees almost perfectly with the FLUENT results, if a constant value for the Sherwood number is used. Such correlations are especially useful for computer codes like RAFT, VICTORIA and MELCOR that can not use much computing effort for single phenomena.

A TGT apparatus, based on the one used for these radio-tracer experiments, will also be used for revaporization experiments of the radioactive Phebus FPT-1 samples. The furnace is reduced in size to avoid excessive lead shielding weight and to make handling of the process tube in a glove box practicable. However the experimental conditions like gas composition, gas flow rates and temperature ramp rate for TGT tests with the Phebus FPT-1 samples remain as close as possible to those described in this paper.

4. LWR aerosol experiments at VICTORIA model containment

The most important conditions affecting the aerosol behaviour in the containment are atmospheric relative humidity, and temperature gradients at structure surfaces. For particles of inert materials the problem is fairly simple, since they grow by steam condensation only in saturated conditions. However, a substantial part of the airborne radioactive material in a severe reactor accident is hygroscopic. Hygroscopic particles absorb water also in superheated conditions. The experimental results obtained in the VICTORIA facility clearly demonstrate the effect of hygroscopicity causing strong particle growth and settling at saturated conditions.

The risk and consequences of a hypothetical severe reactor accident depend much on fission product aerosol behaviour in the containment. In the course of such an accident radioactive fission products, structural and control rod materials are released from the core in vapor and aerosol form. Most of the vapor is rapidly converted to aerosol particles due to gas cooling and chemical reactions. Only noble gases and part of iodine can exist as gaseous species. The form of iodine is plant and accident sequence specific. During the transport through the RCS (Reactor Coolant System) part of the aerosol particles and vapors is deposited on the surfaces and part is released into the containment. Other aerosol sources to the containment are molten core-concrete interaction after the vessel melthrough, high pressure melt ejection in a high pressure case, revaporisation of deposited material from RCS surfaces and re-entrainment from boiling or flashing water pools.

In the case of a containment failure or pre-existing leakage the source term is directly related to the amount of airborne radioactive matter in the containment atmosphere. One important factor affecting the concentration of containment aerosols is the evolution of particle size as a function of time. Particle size is changed due to coagulation and steam condensation on particles. The growth is enhanced, if the aerosol released from the core contains hygroscopic material (e.g. CsOH, $CsBO_2$, Cs_2CrO_4, CdI_2, CsI...), which absorbs water below 100 % relative humidity. Under favorable conditions particles may grow to sizes larger than 10 µm. These particles will settle rapidly, after which only small fraction of micron-sized (or smaller) particles remain airborne. For accurate prediction of containment aerosol behaviour also the correct modelling of thermal-hydraulic conditions is a crucial factor. Thermal-hydraulics have a direct impact on steam condensation on particles through the prediction of relative humidity and temperatures and on deposition through the predicted flow patterns.

The engineering safety features (sprays, ice condensers, pool scrubbing), if available, seem to be very effective in removing aerosols from the containment atmosphere. Today also Filtered Venting Containment Systems are widely used as a measure for preventing unacceptable high source terms from severe core melt accidents. The efficiency of the engineering safety systems is, however, sensitive to the aerosol properties to be removed. Thus it is important that we are able to predict the chemical composition, particle size and mass concentration of the radioactive and in-active materials in the containment atmosphere as well as possible.

The aim of aerosol experiments in the VICTORIA facility is to validate the containment aerosol models used in the nuclear reactor accident codes (Mäkynen et al., 1998). As a final goal we need to confirm that containment aerosol codes are able to calculate correctly the radioactive hygroscopic and non-hygroscopic aerosol behaviour in non-homogeneous multicompartment containments. It is possible to reach this goal by making experiments in the well controlled and instrumented model containment (VICTORIA multicompartment test facility, Loviisa Nuclear power station model containment with linear scaling 1:15). Earlier a research programme has been carried out in this facility to study the TH behaviour and hydrogen distribution in severe accident conditions.

The results of these aerosol tests will extend our knowledge from single compartment test facilities to multicompartment non-homogeneous TH conditions. Large scale experiments have indicated that in the containment there are large local differences in the temperatures, flow fields and relative humidities. These differences affect significantly the containment aerosol behaviour.

4.1 Experimental and modelling

In the quantitative containment aerosol behavior tests at the VICTORIA facility state-of-the-art aerosol measurement systems are used. Aerosol number and mass concentration is measured continuously using Condensation Nucleus Counter (CNC) and Tapered Element Oscillating Microbalance (TEOM) mass monitor. Particle mass and chemical composition size distributions are determined by Berner Low Pressure Impactors (BLPI). Measurements have also been made with Electrical Low Pressure Impactor (ELPI), filter samplers, deposition coupons and deposition trays.

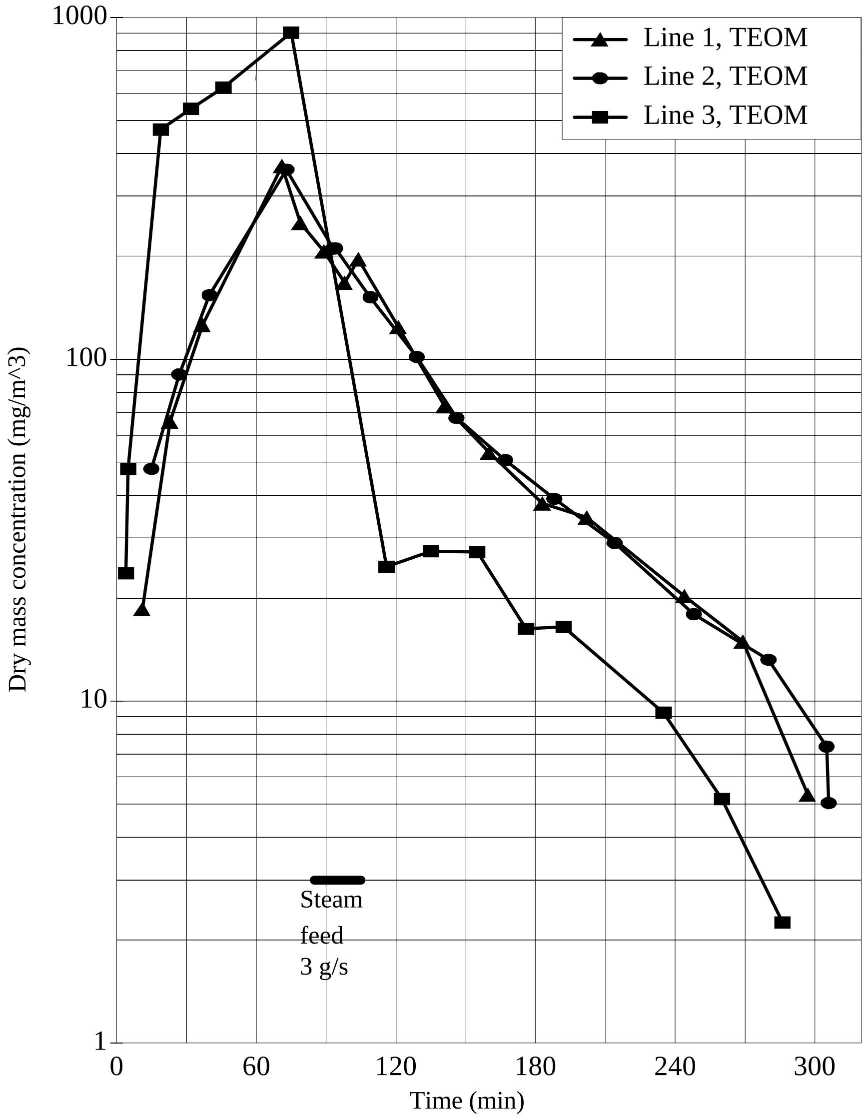

Figure 5. CsOH mass concentration as a function of time. The inputs of the lines 1 and 2 are in the upper compartment and the input of the line 3 is in the lower compartment.

The TH-data sampling system measures and records flow rates, pressures, temperatures and RH-values in the system. Temperatures are measured with T-type detectors. RH-values and temperatures are measured with several normal

VAISALA humicap detectors and with two (upper and lower compartment) new heated type VAISALA dewpoint detectors. In these new detectors RH-sampling head is temperature controlled which prevents water condensation into the detector head even when RH is over 100 %.

At the moment 12 experiments have been performed (5 TH-tests and 7 aerosol tests) Used aerosol materials have been NaOH, CsOH and Ag. Experimental results have also been compared with the severe accident codes CONTAIN and FIPLOC (Mäkynen et al., 1998). An example of the results is shown in figure 5 where the dry mass concentration of CsOH is presented as a function of time.

FIPLOC and CONTAIN are both so-called lumped-parameter containment codes, in which the containment is modelled as a number of homogenised control volumes, interconnected by junctions.

FIPLOC is a part of the comprehensive mechanistic containment code system COCOSYS. It includes the RALOC containment model for the calculation of thermal hydraulic conditions. The CONTAIN code has been developed by USNRC severe accident research program for predicting the physical, chemical, and radiological conditions inside the containment in severe accidents.

The MAEROS aerosol model, which takes into account the effect of aerosol material hygroscopicity, is applied in both codes. The code versions used in the calculation of the VICTORIA experiments were FIPLOC 3.1 and CONTAIN 1.2.

5. Discussion

Because the effect of hygroscopicity on the particle growth depends on the RH of the atmosphere, an accurate TH pattern is essential for any analysis of aerosol behaviour. In a NPP containment, there are numerous compartments and rooms which maintain different TH conditions, even locally inside a large volume. A good example of such a volume is the upper compartment in the Loviisa ICC. While the dome is well mixed by the convective loop flow, there is a stagnation layer at the IC top level, with a strong temperature and humidity gradient between the dome and the lower dead-ended volume. This kind of a strong gradient between two inter-connected volumes cannot be modelled with a lumped-parameter code.

The flow patterns in the containment are another significant source of deviation in aerosol transport and deposition. Also, considering the large volume of a containment — 60 000 m^3 — it is impossible to create a nodalisation in which the atmosphere of each control volume could be considered homogeneous, and the calculational effort would be reasonable. Inaccurately modelled convective flows, combined with unrealistic TH conditions in even a small part of the multicompartment system, can lead to significant errors in the overall aerosol modelling.

The MAEROS aerosol model requires input values for the density and solubility of aerosol materials, and the initial particle size distribution. The same density is used for all aerosol species, and is constant throughout the calculation. In the modelling of complex problems with various aerosol species of different properties, such a global parameter cannot be defined. Even the behaviour of such a single aerosol species, the hygroscopicity of which depends on the surrounding conditions, can be problematic. And even more so, if the surrounding TH conditions are changing.

The aerosol parameters mentioned above are just examples of the numerous values to be defined by the user for an input deck of a containment code. User-made choices in nodalisation as well as problem description can result in significant deviations in the final results. International benchmark exercises have been organised to compare the outcome of several code users on a specific problem.

Also, the characteristics of the modelled facility determine which phenomena play the most significant role in the end results. For modelling aerosol behaviour in the VICTORIA small scale facility, correct estimation of the particle size is important because of a relatively small deposition height. However, the most significant source of uncertainty (regarding atmospheric aerosol concentration) in the full-scale modelling of the Loviisa containment can be expected to be the loop convection, and its decontamination efficiency. Furthermore, in containments with less effective atmospheric turnover, the particle size becomes important again.

An issue that questions the comprehension of sensitivity analysis and code verification by other codes is the fact that so many containment codes use the same aerosol model MAEROS. On the other hand, the work presented here indicates that most of the straits in containment aerosol modelling can be traced not to the aerosol model package itself, but to its physical and numerical coupling to the thermal hydraulic calculation, and parameter choices.

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Risk and reliability analyses (LURI) and expert judgement techniques

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1. Introduction

Probabilistic safety analysis (PSA) is currently used as a regulatory licensing tool in risk informed and plant performance based regulation. More often also utility safety improvements are based on PSA calculations as one criterion. PSA attempts to comprehensively identify all important risk contributors, compare them with each other, assess the safety level and suggest improvements based on its findings. The strength of PSA is that it is capable to provide decision makers with numerical estimates of risks. This makes decision making 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. For example, the PSA scope has been limited to the power operation and process internal events (transients and LOCAs). Only lately, areas such as shutdown, external events and severe accidents have been included in PSA models in many countries.

Problems related to modelling are, e.g., that rather static fault and event tree models are commonly used in PSA to model dynamic event sequences. Even if a valid model may be generated, there may not be any other data sources to be used than expert judgement. Furthermore, there are a variety of different techniques for human reliability assessment (HRA) giving varying results.

In the project Reliability and Risk Analyses (LURI), these limitations and shortcomings have been studied. In the decision making area, case studies on the application of decision analysis and a doctoral thesis have been published (Holmberg, 1997). Further, practical aid has been given to utilities and regulatory decision making. Model uncertainty effect on PSA results has been demonstrated by two case studies (Pulkkinen & Huovinen, 1996). Human reliability has been studied both in the integrated safety analysis study (Holmberg et al. 1998) and in the study of maintenance originated NPP component faults based on the maintenance data (Laakso et al. 1997). The former has generated a rather unique approach between psychologists and reliability engineers, whereas the latter study reveals important aspects of maintenance related human errors such as the role of instrumentation and outages. Expert judgement has been one of the major themes of the project throughout its life and will be discussed more in the following.

2. Treatment of PSA modelling and data problems by using expert judgement

2.1 General

The most well known quantitative expert judgement elicitation techniques are so-called direct (numerical) elicitation and indirect elicitation. For example, paired comparisons (e.g. Bradley 1953, Torgerson 1958) and various ranking and rating techniques (e.g. Saaty 1980) belong to indirect elicitation techniques. The combination of judgements has normally taken place either by using group consensus techniques or by mathematical aggregation. The group consensus techniques, such as Delphi and nominal group technique are discussed e.g. by Dalkey (1969) and Delbecq et al. (1975). Mathematical aggregation has taken place e.g. through simple averaging, through expert determined weights e.g. DeGroot (1974) or by Bayesian methods e.g. Mosleh and Apostolakis (1984) and Pulkkinen (1994).

The actors involved in the expert judgement are a decision maker that has a problem to solve, normative experts who deal with the method and substance matter experts who are familiar with the problem or with its parts. In brief, the expert judgement means that normative experts elicit the knowledge of substance matter experts and combine it to form a basis for the decision making (e.g. Cooke 1991).

The elicitation process is important in order to avoid biases. Biases may be caused by social pressures in the elicitation situation, over-reliance on own knowledge and using simple heuristics to determine the judgements. Examples of such heuristics are anchoring, i.e. adjusting the first value only little although new evidence would say something else, and availability, i.e. experienced phenomena get larger value than they deserve (Tversky & Kahneman, 1974). Other bias types are known as base rate fallacy and overconfidence dealing with problems in probability assessment. Otway & von Winterfeldt (1992) also discuss motivational and structural biases - the first one results from assessor's dependence on the topic to be evaluated and the second to framing. For example, the probabilistic results look very different on a lognormal scale than on the linear scale.

Due to potential biases, several requirements have been set in the literature to ensure proper use of expert judgement, e.g. in (NRC 1990, Cooke, 1991). First, only experts that have demonstrated they expertise should be selected. Then the analysis should be reproducible, accountable, subject to empirical control, neutral and fair. Here, reproducibility means that all the calculations and analyses have to allow to be traced back to their origin and repeated. Accountability means that the values given can be traced to their source (expert). Empirical control means that the results should, in principle, allow them to be falsified by empirical tests etc.. Neutrality means that experts should not be able to play with their judgements but instead to be encouraged to give their true assessments. Finally, fairness requires that all experts are treated equally during the analysis, although some scoring rules and weighting procedures might classify them afterwards.

2.2 VTT's expert judgement methodology

In the LURI-project, a new methodology for using and eliciting expert judgements was developed. The methodology is based on that applied in the NUREG-1150 study. As it can be seen from Table 1, the phases of the method correspond to those of the NUREG-1150 methodology.

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. Another 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.

Table 1. The phases of an expert judgement process.

Phase	Description
1. Selection and training of experts	• expert selection criteria: demonstrated expertise, relevant education and work experience • training topics: concepts of probability theory and statistics, the expert judgement methodology, calibration, nature of subjective probability statements
2. Elicitation of expert judgements	• case dependent • fractiles of distributions (preferable form) • direct estimates or paired comparisons • two phased procedure: 1) initial estimates, 2) individual expert analyses, expert discussions, final estimates
3. Modelling and combination of expert judgements	• application of a hierarchic Bayesian framework • calculations by Monte-Carlo methods • comparison to direct aggregation results
4. Sensitivity analyses	• e.g. with respect to the number of experts, individual expert statements and to the ability to truly give the fractiles.
5. Discussion and feedback from experts	• review of the combined distributions and thinking models
6. Documentation	• experts' reports, • final report

The training includes discussions on the concepts of probability theory and statistics. If necessary, the concepts are illustrated by simple examples. Then, the used 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 may be a difficult task. In some cases, it is enough that the experts directly give estimates for the unknown variables, sometimes they are also asked to assess their uncertainty about the estimates - or to give the 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 LURI project, the experts are allowed to discuss about the issue, but numerical estimates are not 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 to the case under consideration is discussed. The normative experts lead the discussion.

The experts give their assessments individually after the discussions. The assessments are made and documented following issue specific formats, 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 combination of the assessments takes place in the VTT method by using a Bayesian approach (e.g. Pulkkinen & Pyy, 1996). The Bayesian modelling framework consists of describing the full distribution of all random variables in the model, and including the expert judgements as a Bayes network. The Bayes network corresponding to the model applied in this study is shown in Figure 1. The variable of interest is denoted by X. We assume that the uncertainty about the value of X can be represented by a Gaussian distribution, or its transformation such as lognormal or logit, with parameters μ and σ with unknown values. The uncertainty concerning the parameters is represented by non-informative prior distributions (see e.g. Gelman et al. 1996, Box & Tiao, 1972). Experts, the number of which is *m*, are asked to express their uncertainty on *X* with selected percentiles. The expert judgements, denoted by $\boldsymbol{Y_i} = (Y_{j0.05}, Y_{j0.50}, Y_{j0.95})$, where $Y_{j\alpha}$ is the α-percentile given by the expert j, are related to the parameters μ and σ through a set of hidden variables $\Theta_{11,}$, Θ_{12},..., Θ_{1L},..., Θ_{mL}. These variables are assumed to be conditionally independent and identically distributed (given μ and σ) and their distributions is the similar to that of *X*. In other words, the set $\{X, \Theta_{11,}, \Theta_{12},..., \Theta_{1L},..., \Theta_{mL}\}$ consist of exchangeable random variables.

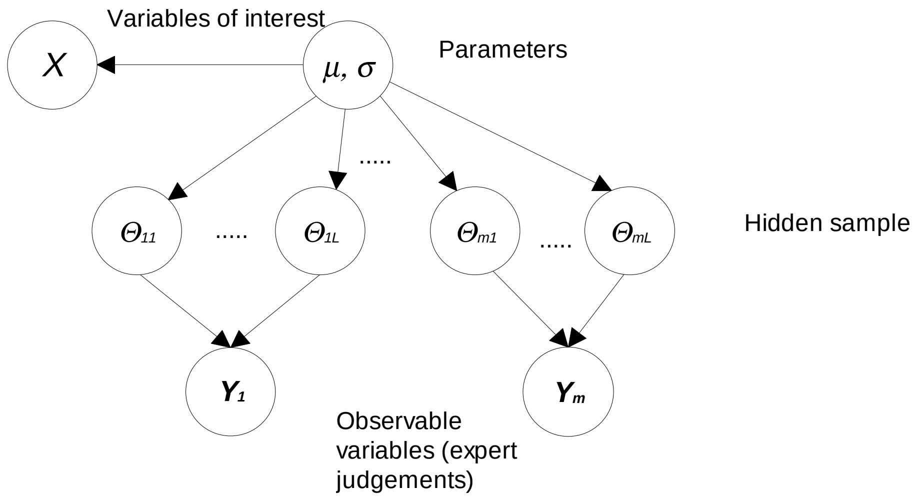

Figure 1. Bayes network for the expert judgement model, notation Yi = (Yj0.05, Yj0.50, Yj0.95).

The percentiles given by expert *j* are interpreted as sample percentiles of the hidden sample of expert j. However, the values of the variables in the hidden sample $\{\Theta_{11,}, \Theta_{12},..., \Theta_{1L}\}$ remain unknown, and the experts' percentiles specify this sample only partially. Thus, we may interpret the role of the hidden variables as follows: the experts have in mind a sample, of *X* e.g. based on history or on a model, of similar variables, but they are not able to specify the sample perfectly. Since the hidden variables depend on the parameters μ and σ, and since the experts percentiles are straightforwardly related to the hidden variables (through sample percentiles), the distribution of the parameters can be updated by using the experts' percentiles. The posterior distribution of μ and σ can be used in determination of the distribution of *X*. In practice, the posterior distributions must be determined by applying numerical methods. The Gibbs sampler (see e.g. Chib & Greenberg 1995, Casella & George 1992) appears to be a good method for determining the posterior predictive distributions for *X*.

2.3 A case study

2.3.1 Description of the case study

The case discussed in this paper is an HRA study for a shutdown LOCA case of Barsebäck NPP. Barsebäck NPP is an ABB delivered BWR with four external main circulation pump loops and shutdown cooling system connected to the loops. The study object was the inadvertent opening of an isolation valve in the shutdown cooling system (pipe diameter 250 mm) and the follow-up balancing actions. The system is located below the reactor core level, and together with opening another disassembled valve, results in a rather rapid core uncovery. The time estimates vary from 30 to 60 minutes.

The domain experts came from the power company and their consultant enterprise. They represented a wide variety of experience areas ranging from behavioural science to reliability engineering, NPP maintenance and operation. The normative experts came from VTT Automation. The expert training was organised in the form of a seminar so that, first, expert judgement techniques, heuristics & biases in their use and human reliability analysis were discussed generally. Then, the VTT expert judgement methodology was discussed. The seminar ended up with a direct case dependent expert training session, where ample time was given for free discussion about the case. The probability for causing the leakage (θ), the time physically available (T1) and the time the balancing action takes (T2) were selected as variables to be elicited.

The judgement elicitation was made in two steps. First, the initial estimates for the variables were given just after the training session, in which the issues and variables were fixed. The experts were asked to give their 5%-, 50%- and 95%-quantiles for the unknown variables. Then, the experts were asked to perform their individual analyses of the case and provide their final estimates followed by the basis for their reasoning in the form of a report.

The reports were presented in the elicitation session. In their presentations, the experts were not allowed to present their quantitative estimates. The main aim of the experts presentations was to discuss and compare the experts' models and approaches. After the experts' presentations, the normative experts elicited the quantitative estimates from each expert individually in an interview. During the interview, also some consistency check-ups took place, e.g. it was controlled whether the assessments really were 5% and 95 % values instead of min and max values.

After the final elicitation interviews, the normative experts performed a tentative combination of the experts' quantitative assessments and draw some preliminary conclusions with regard to their reasoning and modelling of the case. These results were, then, communicated to the experts and comments were received from them, but no expert wanted to change their judgements. Apart from this, a feedback session was arranged as a part of final reporting. A sensitivity analysis was also performed to compare our Bayesian approach to the direct comparison of the experts' judgements and the effect of neglecting a single expert at a time.

2.3.2 Results of the case study

The experts' reasoning with regard to the time to core uncovery situation (T1) was rather straightforward. Actually, two first elicitation round judgements gave nearly one point value for the cumulative distribution fractiles, showing a high confidence on previous analysis work. The situation changed in the course of the analysis, since the contradictory numbers and text in different analyses were discussed leading to an increased uncertainty in the values. Explanations to the increased uncertainty could be based on the fact, if the pool water inventory or the pipe friction had been taken into account, or not, in the calculations.

All the experts took the probability of a failing electrical closure of the valve, leading to the need to close it by hand wheel, implicitly into account in the time to balance the situation (T2). In their first round evaluations, some of the experts considered whether it is even possible to enter the room to close the valve manually. This led them to overly conservative 95 % percentiles, first, that then decreased in the elicitation session. In some analyses, also a partial closing of the valve with a successful core spray injection was taken into account. On the other hand, the awareness of the experienced difficulties to remotely steer components in shutdown conditions affected the distributions. The difficulties are due to the amount of overhaul and modifications in electrical and instrumentation systems during a refuelling outage.

The treatment of the probability of leakage θ, in experts' reports and interviews, was based on two structured models of the leakage initiation, one of them with several routes that could generate the leakage. In addition, two history data based quantification methods and one more philosophical discussion about probability concept and uncertainty were given. The history models were based on the fact that this kind of a leakage has taken place in the BWR reactors in the whole world. This led to the calculation of the total cumulative yearly experience of the world's BWR outages. Generally, plant barriers were regarded as well functioning. On the other hand, transgressions of work permits and of safety umbrellas take place in almost every refuelling outage.

In the experts' analyses, a wrongly timed test, carried out from the valve switch gear unit dominates other leak causes. Other potential mechanisms are short circuit or test from the control room with failures in the safety umbrella package, such as fuses left in their place. The risk dominance of a test, taking place outside the control room, is due to the fact that it is carried out without a specific work permit and according to a long list of electrical limit switch tests in outages.

As a general observation about experts' estimates, all of them changed their first judgements. Thus, we could claim that, at least, a total anchoring could be avoided. The time distributions for T1 and T2 became closer to each other. This is mostly due to the decrease in conservatism regarding T2 and in overconfidence regarding T1. A more thorough modelling and analysis of historical evidence led to the smaller probability distribution values for θ. The evidence showed that neither group work had taken place, nor that the experts had intentionally arrived in lower estimates than in their first judgements.

The summary of the experts' combined distributions is shown in Table 2. For T1 and T2, the calculation was based on lognormal trans-formation and for θ on logit transformation. It is important to note that for θ, the 5 % and 50 % values are rather low but the long tail, due to the skewness of the distribution, leads to rather high mean and 95 % values.

Table 2. Summary of the combined distributions.

	The posterior means and 5%, 50% and 95% quantiles of the variables											
	Variable 1: T_1 = Time to core uncovery", min				**Variable 2: T_2 = "Time needed to balance the situation," min**				**Variable 3: θ = "Probability of the leakage"**			
	5%	*50%*	*95%*	*Mean*	*5%*	*50%*	*95%*	*Mean*	*5%*	*50%*	*95%*	*Mean*
Initial estimates	24.5	35.0	49.0	35.7	1.4	16.0	175.0	60.8	4.6E-6	1.2E-3	1.8E-1	4.2E-2
Final estimates	20.0	40.0	79.0	40.2	3.6	21.5	136.0	41.8	6.0E-9	2.6E-5	6.0E-2	3.6E-2

The core uncovery event can be presented as a combination of two events: A1 ={T2>T1} = " the time needed to balance the situation (T2 } is longer than the time to core uncovery" and A2 = "a leakage occurs" . The probability of uncovery may be calculated from formula (1):

$$P(A) = P(A_1 | A_2) P(A_2) = P\left[(T_2 > T_1) | A_2\right] P(A_2) \qquad (1)$$

Since the mean values may be used in the calculation, the probability of the accident is a single number, not a random variable with a distribution. This result leans on the Bayesian interpretation of the model, and it is compatible with the "integrated uncertainty analysis" concept discussed e.g. by Pörn & Shen 1992. The Monte-Carlo simulation (1000 rounds) of the distributions produced the expectation 2.3E-3 for P(A).

Sensitivity analyses were performed to check the sensitivity of the results to individual values and to the combination model used - Bayesian versus arithmetic averaging. No large differences were obtained for variables 1 and 2, but for variable 3 there was some sensitivity. The reason for the differences is the fact that it is not easy to fit a parametric distribution with only three fractiles. Further, in our case, the experts fractiles did not always correspond well to those of lognormal or logit distributions that we used. Generally, the direct combination produced some narrower distributions. The results for the total probability of the core upper grid uncovery would remain somewhat lower by using direct combination than those by using the Bayesian approach.

2.5 Other applications of VTT's methodology

The case described earlier is only one example of the applications of VTT's methodology. It was chosen as an example because human reliability analysis (HRA) has been one of the themes of LURI. It is clear that expert judgement will always play important role in HRA. Apart from HRA, VTT's expert judgement method has been used in several applications, e.g. in EU Bechmark Exercise for expert judgement techniques (Cojazzi 1996). Applications of VTT's method are shown in Table 3.

Table 3. Some applications of VTT's expert judgement method

Case	**Elicited variables**	**Type of elicitation**	**Type of combination**
Application to the results of a fuel-coolant interaction experiment (EU)	Physical variables	Three fractiles (5%, 50%, 95%)	Bayesian combination (VTT method)
Application to the analysis of the hydrogen issue (EU)	Event probabilities	Three fractiles (5%, 50%, 95%)	Bayesian combination
Barsebäck shutdown LOCA (NKS)	Probabilities and time windows	Three fractiles (5%, 50%, 95%)	Bayesian and direct combination
Heavy load transport	Probabilities	Three fractiles (5%, 50%, 95%)	Bayesian combination
Traffic safety analysis	Relative risk curves	Experts' relative risk curves combined with point estimates	Bayesian combination
Integrated safety analysis (HRA)	Time windows	Three fractiles (5%, 50%, 95%)	Bayesian combination

3 Discussion and conclusions

Expert judgement has proved to be a valuable to in collecting, combining and presenting information on topics for which data is otherwise sparse. PSA and availability studies and their applications like reliability centered maintenance require so much different kinds of data that it cannot be ever collected in the form of purely objective statistics. Actually that kind of statistics seldom exist – they are affected by subjective decisions in the drafting phase and they always have to be interpreted by humans. Another type of application is the analyses needed in connection of PSA review where independent comparative analyses are needed. If there are not resources to make a full scale analysis on the issue, structured expert judgements provide a flexible tool.

The VTT methodology has shown its worth in practical work. However, expert judgement always requires a more rigid discipline by the practitioners. Methods have to be transparent, scientifically acceptable and help the experts to give their real judgements. The results also have to be understandable in order to form a basis for the decision making.

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Organisations and their safety processes

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Abstract

Organisational factors have in many incidents and accidents proved to be one of the most important contributors to human errors at nuclear power plants (NPP). The problem with this finding is that very few methods exist for the identification of organisational deficiencies which may contribute to high error probabilities. Methods for the support of managing high reliability organisations have been the target of research efforts in VTT Automation. The paper gives a brief reference to some research which has been carried out in connection to the LURI- and ORINT-projects.

1. Introduction

It is widely recognised that safe and reliable operation of high risk industrial systems depends not only on technical excellence, but also on individuals and the organisation. An efficient management of safety and availability implies an in-depth understanding of behavioural characteristics of the operated system. including technological, human and organisational aspects of operation. Systems have grown larger and more complex which has made their control more difficult. One remedy is to design the systems to be better adapted to their users and another is to make managers better aware of couplings between technical and organisational performance.

VTT Automation has been involved in nuclear safety research since the early seventies. The projects have included control systems, human factors, reliability assessments, training simulators and operator training. Partners in the project have been the nuclear power plants (NPP) both in Loviisa and Olkiluoto as well as STUK, the Finnish regulatory body. In addition VTT Automation has been involved in several international project within the Nordic nuclear safety research (NKS) and in co-operation with international organisations. The research has been based on a fruitful combination of systems engineering and behavioural sciences. The close co-operation with the nuclear utilities in Finland and also to some extent in Sweden has provided a fruitful combination of theoretical and practical challenges. Various activities with an application to high reliability organisations have recently been brought together to form a central research theme within VTT Automation.

2. Nordic nuclear safety research

Control room design, human reliability and operator training were selected as major themes in the Nordic nuclear safety research (NKS) when the research was started in 1976. Later programmes addressed among others themes such as organisational influences on human reliability, information support in emergency management and human interactions in PSAs. Several organisations in Denmark, Norway and Sweden have together with VTT Automation been actively engaged in the projects. These organisations include the Risø National Laboratory in Denmark, the OECD Halden Reactor Project in Norway and the Swedish Nuclear Power Inspectorate. The work has over the years contributed to an improved understanding of various safety issues connected to nuclear power.

The last Nordic four year research programme in nuclear safety was completed in 1997. The inputs from VTT Automation to this programme were co-ordinated closely with the two RETU projects LURI "Reliability and risk analyses" and ORINT "Human factors in NPP operations". One of the projects NKS/RAK–1 "Strategies for reactor safety" addressed various safety aspects, including management and organisations. In the subproject NKS/RAK–1.1 "A survey and an evaluation of safety activities in nuclear power" a broad investigation of various safety activities was made. In the subproject NKS/RAK–1.5 "Modernisation for safety in Nordic NPPs" the plant modification processes and the ongoing modernisation projects were studied and assessed. In both subprojects a special consideration was placed on a comparison of safety practices in Finland and Sweden. A large number of people from Finnish and Swedish utilities and regulatory bodies were interviewed as a part of work.

The general impression from the studies is that the safety activities at the NPPs in Finland and Sweden are efficient and well targeted. One problem in the industry is that the work load of people generally is high with occasional peaks of extremely high work load. This problem seems to be partly connected with a high ambition in the work and partly with mounting economic pressures caused by the deregulation of the electricity market in the Nordic contries. Experience from the studies point to the benefit of comparisons of practices as a method to identify important components of organisational safety and possibilities for improvements. The interviews proved to be an efficient way of rapidly acquiring an impression of the performance of an organisation. Specific problems can be followed up by more detailed studies.

3. The organisation seen as a control system

The organisation of a NPP can with an analogy be seen as a control system which ensures that activities and work processes are carried out appropriately and efficiently. This control system is implemented by people and through people, which means that it is both self-structuring and adaptive. Important questions are how organisational structures and work procedures influence the quality of work and which measures are needed to achieve an acceptable level of safety. Various evidence from NPP operations gives an impression that some work processes and practices are more efficient than others. The challenge is to identify and rectify unsafe practices before they result in incidents or accidents.

Control systems rely on the concepts of *set point*, *control action* and *feedback*. Organisations can similarly be said to rely on the concepts of *goals*, *means* and *experience*. Goals have to be defined and further specified according to the line structure of the organisation. Means for achieving specified goals have to be selected among best available practices. Experience has to be collected and used to improve applied practices and to reconsider the goals. To be efficient the considerations of goals, means and experience should be an integral part of each organisational unit of a NPP.

Two other important concepts are *quality* and *barriers*. The quality concept relies on a defined quality and a description of how this quality can be reached. Regular audits to ensure that defined methods are used in the work processes are parts of the quality system. Barriers are used in technical systems to prevent unsafe excursions. Administrative barriers are used in a similar way to prevent work of a low quality to proceed within the system. One important barrier is an independent safety assessment which is carried out both within a NPP by the utility itself and by the regulator as an outside activity.

4. Decision making in NPP organisations

There are many similarities between organisations in general and organisations managing NPPs, but there also important differences. The most important difference is the very high safety requirement which is due to the fact that the reactor requires a continuous attention and that failures in this respect can lead to radiological hazards. The high safety requirement leads to conservatism and reliance on proven solutions. A NPP is very complex system which for its operation demands high skills in several different disciplines. The complexity of the interaction between various technical systems on one hand and between the technical systems and the human and organisational systems on the other hand, makes it very difficult to predict in detail how a NPP will behave in a specific situation.

Hands on operational decisions are made in the main control room. The control room operators depend on convenient information presentations which support the detection of deviations in main plant variables and the selection of correct actions for handling disturbances. Preventive and corrective maintenance is carried out according to plans and in response to failures. The yearly planning process identifies plant modifications to be implemented within the planning period of operation and outages. Strategic planning has a longer time horizon in which needs for investments are considered in the context of both technical improvements and development of human resources.

Models of a rational decision making are based on expected utility theory. According to the theory the utility of an outcome of a decision is weighted with the probability of achieving that outcome. The normative decision rule is to select the option which gives the highest expected utility. The theory often gives useful insights, but a strict normative approach is seldom practical in real world decisions. The problem is that in practice it is not possible elicit individual utility functions and obtain probability estimates with a necessary accuracy. In real time decision making the theoretical framework has to be reduced to simple rules of thumb to be practical. The consideration and resolution of conflicting objectives pose their own problems.

5. Components of organisational excellence

A definition of organisational excellence has to rely on some model. The problem in finding a suitable model is to balance between a model which is too simple to give only trivial answers and a model which is too complex to be practical. The model should also be understandable for those who are expected to use it. A model can be used to help in collecting information on the performance of an organisation as well as for improvements. Before a model can be used it has to be validated.

A model consisting of six basic principles important for an efficient management of safety has been suggested. The principles can be described by the concepts of *organisation, planning*, *models*, *information*, *feedback* and *priorities* which reflect interconnected resources and activities. An organisation is built through the allocation of responsibility and authority which should be balanced, structured and described. Systematic planning including a definition of goals and an allocation of resources is a prerequisite for good performance. Efficient concepts, models and processes should be utilised to support safety activities. Correct and timely information in all operational situations and work processes requires an efficient information system. Systematic feedback and use of operational experience for organisational learning is crucial for a continuing improvement of performance. Setting of priorities requires methods for assessing costs and benefits of various actions.

The six basic principles can be used to generate an idealised organisational model of activities to be reviewed. This model can in a second step be used to formulate questions by which assessments of an activity can be obtained. In a first set of questions a check can be performed that all important components are covered in the activity. In a second set of questions the functional efficiency of these parts can be investigated. A third set of questions can interrogate the interactions of the components and the efficiency of the interactions. A final set of question is concerned with the documentation of the activity and its parts.

6. Performance shaping factors

Performance shaping factors have been used in PSAs to predict the likelihood of human errors. Performance shaping factors can be used more generally to understand situations where decision makers make their decisions. It is evident that stress and a heavy work load will make decisions more vulnerable for errors. The difficulty in using performance shaping factors is that their actual influence for instance on error probabilities is difficult to assess, but they can still give clear indications for areas where improvements are needed.

Performance shaping factors can be divided e.g. into environmental factors, task related factors, individual factors and organisational factors. Among the environmental factors are things like noise, temperature, radiation, etc. Task related factors are concerned with availability and quality of information and instructions, time for task, time of the day, etc. Individual factors include skills, competency, attitudes, beliefs, etc. Organisational factors include authority, responsibility, structure, communication, etc. Performance shaping factors can be assessed in various ways using both objective and subjective measuring methods.

The influence of unfavourable performance shaping factors can be reduced in several ways. The influence of environmental factors can be reduced by shields and protecting wears or the task in consideration may be automated in parts or completely. Task related factors can be influenced by a re-allocation of resources, by improving man-machine interfaces, by re-writing procedures, introducing various support systems, etc. Individual factors can be influenced by selection and training of personnel. Organisational factors are determined by management styles and practices. Already an awareness of the influence of various performance shaping factors on human reliability can, together with an understanding of the criticality of certain decisions, give a good list of safety related topics that deserve special attention.

7. Performance indicators

Performance indicators are used when performance itself is too complex to be measured directly. Performance indicators are also valuable where actual performance builds on several interdependent characteristics inter-connected by long time constants. Performance indicators can in a way be said to provide feedback on future performance before trends can be seen in actual performance. NPPs use various systems of performance indicators which include mainly "hard" technical indicators. In the consideration of organisational factors the possibility to define "soft" performance indicators, i.e. indicators suitable for assessing individual commitment and organisational efficiency is an interesting possibility..

The definition of performance indicators has to build on actual and assumed relationship between real performance and the suggested indicators. Sometimes it may be difficult to establish a valid causal coupling between an indicator and performance, but that does not necessarily render the indicator useless. Performance indicators should be accepted within the organisation, they should be difficult to deceive, they should reflect true performance and they should be changed whenever needed. A continuous recording of performance indicators can help in detecting weak signals of organisational deterioration. The benefit of well defined performance indicators is that they make it easier to set priorities through the quantification of important aspects of safe operation. Performance indicators can to some extent be used to compare the efficiency of various parts of an organisation. Performance indicators are well suited to normal situations, but they are not adapted to an analysis of incidents. Depending on the indicator, the necessary data for its calculation can be collected either by continuous measurements or by taking samples at regular intervals.

One way of taking the full advantage of performance indicators is to integrate them into the management processes. They can and ought to be incorporated into the process of defining goals and targets and they should form the backbone of the feedback of experience. Ideally performance indicators should be linked to company values which can give them a valuable position in internal discussions on goals and priorities. Performance indicators are sometimes used to create additional incentives within the organisation through a bonus system. The indicator system should be the object of a continuous assessment and redefinition to ensure that real performance, and not the indicators, is controlling the path of organisational development.

8. Benchmarking safety

A comparison of work practices, a benchmark, can give valuable information on different ways of designing and conducting safety related activities. A benchmark builds on some similarity between the work processes compared, but it also benefits from the differences. If differences are found in the practices, the question is which practices are better and why. A benchmark can be brought to a quantitative level e.g. by comparing the allocation of resources and the efficiency of concerned work processes.

A benchmark carries several difficulties. Practices may be difficult to compare, because they are too different. Findings may also be difficult to transfer between different cultural settings, because they may have been produced by different socio-economic environments and/or legislative systems. There is also an inherent expectation in a comparison that one of the practices is the best, which is not necessarily true, because each practice has evolved in a process of adaptation to a specific situation. Nevertheless, observed differences have to be interpreted and understood in detail before conclusions can be drawn.

One issue in every benchmark exercise is to identify topics to be addressed. A second issue is to agree on the depth of the exercise. A decision to go deeply into the activities may involve a large effort, but a shallow study may not bring up the important questions. Activities with a large influence on safety are always more important to investigate. For a benchmark between several organisations it is often necessary to involve outsiders to ensure independence in interpretations and recommendations. It may also be easier to achieve the necessary openness in interviews when they are carried out by persons outside the organisations. In the evaluation of impressions from interviews it is necessary to understand that there always are tensions in organisations. The existence of tensions should therefore not be interpreted as an indicator of problems. Such tensions are typically handled within regular management processes and one may even claim that they are an important ingredient in maintaining a continuing safety.

9. Conclusions

The consideration of organisation and management issues as contributors to nuclear safety is becoming increasingly important. One of the difficulties is the absence of a theoretical framework within which organisational factors and their causal relationship can be handled. Such an theoretical framework could also support data collection and organisational development. In a comparison with the technical systems it seems evident that new modelling methods for organisation and management will be required.

Safety is a fundamental prerequisite for the use of nuclear power. The extreme safety requirements of nuclear power will need special precautions and methods. The consideration of high reliability organisations as an object for research may help in this endeavour. A fruitful combination of theory and practice is a necessary precondition for success. A close interaction between systems and behavioural sciences is another where VTT Automation has been involved. If these efforts will succeed nuclear power can remain as a realistic energy option also in the future.

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Evaluation and development of process operators' working practices

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Abstract

The practical aim of our research was to enhance the safety of NPP operations through the development of competencies and design of man-machine interfaces, and through contributing to safety management by providing better human reliability assessment methods. A prerequisite for achievements in these issues is understanding of the nature of the work in the NPP. We have focused on the comprehension of the control room operators' core task. With the premise of the intentional nature of human activity we have developed a new contextual approach for the analysis of activity in real-life situations. It is called the Contextual Analysis of Working Practices (CAWP). Habit of action is a central concept, and we have proposed a practical way to identify habits of action through the analysis of the actors' ways of taking account of the possibilities and constraints of the situation and of using available resources.

We have carried out empirical studies in two nuclear power plants and executed four series of simulator experiments. This has taken place in close co-operation with the simulator trainers and experts of the plants, and nearly all control room crews of these plants have been involved. The central result of this work is the development of the CAWP methodology. With the help of it we have identified differences in the NPP operators' working practices that seem to have relevance for the adequacy of process control. We have also found indications of the significance of working practices for a situationally adaptive use of information aids in the control room, which ought to be verified later. Our research method has been adapted for a routinely used simulator training method. Moreover, the methodology has been applied as a tool in the validation of control room information aids, and incorporated into a new dynamic human reliability method (not discussed here).

1. Introduction

We see human activity as an intentional, context-dependent enterprise explained through meanings the actors attach to their activity and their directly observable interactions with the environment (Harré & Gillet 1994). Following these methodological premises we have developed a new approach for the analysis of activity in real situations. We call the methodology Contextual Analysis of Working Practices (CAWP). *Habit of action* is a central concept in our analysis of activity in real situations. As we have pointed out elsewhere (Norros & Klemola submitted) it refers to the different ways adaptive interactions are constructed with the environment. Our use of the habit of action concept is closely related with the notion of habit, which has two outstanding features, its intentionality and pre-reflective nature (Dewey 1934, Kestenbaum1977, Boisvert 1998). Habits embody the "achievements and victories" of the organism in its attempt to integrate itself with the environment in satisfying ways. As a result of interaction of habit and environment a *situation* is formed that reflects the integration of emotional, practical and intellectual habits. Thus a situation is an interplay of objective and internal conditions (Dewey 1938). A situation is not experienced as an object of cognitive awareness, because habits operate on a level of experience which precedes deliberate, critical positioning of distinct objects of reflection. An object is matter or meaning to man because through habit it has been formed into matter. By studying these objects the schemata of perception and appreciation can be revealed (Bourdieau & Waquant 1992).

We have proposed a practical way to identify habits of action through the analysis of the actor's ways of taking account of the possibilities and constraints afforded in the situation and of using the available resources (Hukki & Norros 1994, Klemola & Norros 1997, Hukki & Norros 1998). The personal sense of the situation is based on the actor's construction of the object of activity, which *orients* the interaction with the environment (Norros 1995, Norros 1996). Due to this ecological function orientation may also change according to the results of activity. Thus, habits of action are not static or stable dispositions. Problems and contingencies in practical actions may lead to reflective thinking as is proposed by Dewey (1933). Also Bourdieu sees the significance of reflection for the promotion of the course of actions in situations where habits appear invalid. Significant in reflection is the subject's effort to become aware of the restrictions of the societally defined schemata of perception and concepts (Bourdieau & Waquant 1992).

Our research methodology has developed as a result of empirical work on three human factors issues in the NPP operations (studies in other domains have also contributed in the development). The three issues are simulator training, validation of control room information systems and human reliability analysis. In this paper we demonstrate the methodology with the help of results from the two first mentioned applications. The everyday term “working practices” is used for the theoretical concept of “habits of action” in general contexts.

2. Analysis of working practices

In the first study we analysed NPP operator crews' task performance in a simulated disturbance situation with the aim to describe the situated construction of actions (Norros & Hukki accepted). Based on our earlier study (Hukki & Norros 1993) we could expect to find differences in the crews' ways of utilising available process information. Our idea was to show that by analysing how the crews make use of process information and operating methods as their resources, it is possible to reveal differences in the crews’ habits of action.

The results of the previous study (Hukki & Norros 1993) gave rise to the hypothesis that an adaptive interaction with the process would require creation of a coherent conception of the process state on a global level and efficient use of situational information of the process state. Both demands could be considered critical in making an appropriate judgement of the behaviour of a complex object, the inherent dynamic relationships of which are not fully known (Ilyenkov 1974, Brunswik 1955, Elstein & Bordage 1988). These demands were considered essential in the control of the NPP. Thus, they would characterise the core task of the operators’ work, and their fulfilment would be relevant in all operational circumstances. Consequently these characteristics could be used as evaluation categories in distinguishing between different ways of interacting with the process, and, thus, express different habits of action.

In the analysis of the courses of action the utilisation of process information, different operational means and the procedures were used as analysis units. For making the courses of actions intelligible we used particular pre-prepared descriptions of the disturbance situation (reference models). They provided insight in the situation specific safety critical demands of the process, and the constraints and possibilities available for the operators in this context (see Hukki in this volume). The results of analysis revealed two distinct habits of action (Norros & Hukki accepted). In the first one, diagnostic actions were integrated with attempts to stabilise the process. The operational methods that were used expressed attention at the particular situational constraints. This became possible because the crews seemed to pursue towards an interpretation of the nature of situation, and the chain of events that had led to it, with the help of active search for information and shared process dynamic reasoning. We named the emerging pattern of interactions with the process *the interpretative habit of action*.

Another pattern of interactions was also identified. Typical to it was that operators prioritised operational actions in the initial phase of the event. The methods used for the stabilisation were standard ones. The crews did not seem to search for diagnostically critical information, nor did they, in their mutual communication, express attempts to pursue an interpretation of the particular disturbance. This pattern of interaction with the process was named *the procedural habit of action*.

These two habits of action that emerged from the material elaborated and completed our assumptions concerning the NPP process operator's core task. The concrete criteria used for distinguishing these two habits of action were search for information, existence of diagnostic inferences, type of operational methods, interaction with diagnostic inferences and measures for stabilisation, and communication of inferences within the team. They express a more or less clear tendency towards a *coherent* and *situation-specific* interpretation of the behaviour of the process. These characteristics were used as evaluation categories for identifying habits of action, because they were considered to express the adaptivity and appropriateness of the operators' interactions with the environment.

After these preliminary results of habits of action we turned back to the material. We also executed two further simulator experiments with the aim to elaborate the concrete criteria for identifying habits of action. These tasks were carried out in co-operation with the simulator training centre of one NPP with the further aim to design a complete method for the evaluation of the operators' performance during simulator training (Hukki and Norros in press). Through providing means for the conceptualisation of the crew's decision making in context the developed method brings about the essential features of activity in relation to the demands and boundary conditions set by the situation. It provides concrete, situation specific criteria for the crews' task performance and habits of action. The method is aimed to enhance the trainees' competence. The idea is that by helping to make the trainees' bases of inference more explicit the method promotes reflectivity and learning. The conceptualisation of the task situations (see Hukki in this volume) does not only provide bases for the self-reflection among the operators but it also invites the instructors to explicate the criteria they consider relevant for learning situations. Therefore, the method also promotes the instructors' work and competencies.

3. Working practices and the operators' utilization of control room information systems

The typical tools in the modern process control industry are the information systems. Essential to their use is the mediated character of the information they provide of the process (Zuboff 1988). The appearance of the process mainly in alpha-numerical form puts demands on intellectual abilities such as interpretation and judgement. Moreover, through its nature of being a man-made artefact an information system is also a model of the reality. Due to the huge amount of information available of e.g. the NPP process further conceptualisation is necessary in the design of man-machine interfaces. A model of the process is realized during design through selecting and structuring information.

Because man does not only create tools but is himself also shaped by the tools he uses (Heinämaa & Tuomi 1989), it is necessary to reflect on the effects of the tools on the perception of the environment. This applies also to the NPP domain, where it is very important to know how information tools shape the process operators' schemata of perception, and do the tools promote control of the process including learning the dynamics of the process. Studies carried out for validation of control room information systems aim to answer these questions. However, because it is difficult to find out the significance of such systems for perception of the process, these studies often restrict themselves to measurement of the effect of the system on some external performance criterion.

We had a chance to develop a more profound validation concept when invited to study the usability of a particular information system that was designed to aid NPP operators' management of disturbance situations. The point was to use our concept of habit of action as a tool for the analysis of the significance of the different information conditions for the management of the process. We also evaluated the effect of the use of the tool on the performance, but we attempted to make the criteria for adequacy of process control comprehensive and, particularly, context-dependent. As a result, we created the opportunity to investigate the relationship between habit of action and adequacy of process control, which had not been analysed in our earlier studies. According to the experimental design we investigated 6 crews' performance in four different simulated disturbance scenarios, out of which half was managed with and half without the extra information aid (with necessary permutations in the ordering of experimental conditions).

According to our methodology process control was measured with the help of the following criteria: identification of the process state, stabilisation of the process, identification of the causes for the disturbance, redefinition of the production goals. The criteria used for assessing habits of action were: a) way of decision-making (global view of the situation, understanding of the character of the disturbance, taking account of situational constraints and action possibilities); b) way of co-operating (a shared interpretation of the situation and coherent team performance; and c) operator's personal way of coping with problem situations (reorienting in the problem situation, critical evaluation of own resources). Altogether 34 items were used for the assessment of the habits of action.

The analysis provided results concerning the operators' habits of action and the significance of the information aid for managing disturbance situations (Norros, Holmberg, Hukki & Nuutinen, 1997 unpublished). It was found out that habits of action correlated in general with the adequacy of process control. In particular the chief supervisor's management style explained variance in process control. However, from a test-technical point of view only the connections between the two main criteria, way of co-operating and managing problem situations may be meaningful. The situational anchors for the third criterion, way of decision-making, correlate internally with the criteria for adequacy of process control. The latter does not, however, mean that way of decision-making and adequacy of performance were not related. Instead, it seems to bring forward the fact that in a complicated performance - even on a simulator, to say nothing about performance in real-life situations - the habits of action define the task and the situationally relevant evaluation criteria for adequacy of actions. If we choose very general process-related success criteria (e.g. minimum level of the reactor tank) they do not distinguish between expert operators. If, again, more specific criteria are used (e.g. level of tank when making the decision to start pumping) the criterion is related with the situational ways of interacting with the process and reflects differences in habits of action. The difficulty of pre-defining success criteria independently of the operators' real actions and , in particular, without attention to the meanings they attach to them, came up when planning the experiments. Due to the above reasons, operating experts of the plant could not easily find agreement of the criteria for adequacy of process control in the scenarios selected for the experiments.

The information aid for process control that was validated in this study provided information of the central process parameters of the plant in an illustrative way. It also included alarms of safety critical functions, which were associated to relevant emergency operating procedures. The usefulness of the system for process control was found to be situation-specific. The use of the system enhanced performance when the disturbance was characterised by a clear disturbance image, it was evolving rapidly and there were no major deficiencies in the availability of process information. In a more diffuse, slowly evolving disturbances and with reduced availability of information this effect was not evident. This result is coherence with what was said above about the nature of information tools. They provide a conceptual generalisation of the process. If the particular, situation-specific behaviour of the process corresponds well with it, the equipment supports the interpretation and control of the situation well. The concrete evidence of this fact is significant particularly because it emphasises the general conditions of the applicability of information aids and, thus, promotes a realistic attitude in their use.

The information aid was perceived by the operators to reduce stress in the disturbance situation. Moreover, the operators' opinions of the advantages of the information system imply that this stress-reducing effect could be related with the effect of the extra aid on building up a coherent conception of the situation and organisation of the co-operative actions in a difficult situation. We did not find statistically verifiable evidence of a interaction of habits of action and information presentation on the adequacy of process control. However, direct qualitative information of the courses of action and the operators' opinions of the advantages and disadvantages of the system imply differences among the shift supervisors' ways of taking the situation into account in his use of the extra information aid for organising the crew's actions. This would mean that a proper use of the potentials of information presentation systems could be enhanced through the development of situationally adequate habits of action. We attempt to verify this assumption in our further studies.

4. Discussion

The practical aim of our research was to enhance the safety of NPP operations through the development of competencies and design of man-machine interfaces, and through contributing to safety management by providing better human reliability assessment methods. A prerequisite for achievements in these issues is adequate understanding of the nature of the work carried out in the NPP. We have focused our research on the comprehension of the core task of the control room operators' work. Our results have shown that creating a coherent and situation-specific interpretation of the process is one of the elements of the core task. It seems, furthermore, that the mediatedness of the information of the process and the concept-rich tools used in the control put particular demands on and constraints for the development of personal, experience-based knowledge of the process. This kind of knowledge has an important role in the management of the process. Acknowledgement of it as an ingredient of the operators' competence is significant also for creating professional self-confidence among the operators. As this person-bound, mainly tacit knowledge is also important for the crew's shared awareness of the situation, its communication becomes an important issue. Use of personal knowledge should not be suppressed by functional or hierarchical division of labour or through the standardised operating methods, the use of which is, of course, an element of professional process control.

We have based our studies on a methodology, which states that the actions of the operators depend both on the objective conditions of the environment (here mainly the process) and the agent's internal conditions and history (e.g. the adopted schemata of perception). The interaction of these two conditions become overt through the habits of action that are actualised in performance. Development of habits to allow a more and more adaptive interaction with the environment is the aim, which can be achieved only through the actors' own reflections on their actions in problematic situations. This helps in the comprehension of the core task and the eventual pressure for change in it. At the present state of our work, we see a need to develop methods for taking the operators' own accounts of the particular performance situations into account even more extensively. We have prevailingly used the, as such necessary descriptions of the process conditions to make the operators' actions intelligible. The operators' own interpretations of the process phenomena and their performance have been less accentuated in our concrete methods, even though the operators' communications during the performance have been utilised as comprehensively as possible in inferring the sense of operators' actions. The use of the simulator as a tool for analysis of situated actions has possibly had an effect here, because then the event can be pre-defined and analysed thoroughly with the help of NPP experts. In other domains, where the studies have taken place in natural task situations, interviews have been used more comprehensively for making the actions of the operators' intelligible (Norros et al in press, Norros & Klemola accepted). In these contexts the knowledge of the operators' habits of action have revealed the objective conditions of the situations, and promoted to understanding of the core task.

Analysis of the natural task situations will become necessary in the NPP domain, too. The primary motive is to achieve more realistic and ecologically valid results, which is a prerequisite for the generalisability of the results in this kind of studies. As we have stated, the actions in real situations reflect integration of intellectual, practical and emotional habits (Dewey 1938). In any constructive action these aspects of habits are integrated, which creates the internal tension into the activity. Mechanical recognition or repetition of operations lack this integrity of different habits. When forming conceptions of the NPP operators' core task all these aspects should be taken into account. Moreover, actions through which competencies are developed should give rise to the realization of the different aspects of habits, because otherwise only mechanical routines will be learned. Training itself must be meaningful activity, it is not mere preparation for real life (Boisvert 1998). Tasks used in the simulator training should be evaluated critically in this respect. For these reasons, we aim to complement our work with studies in natural working situations.

Finally we feel that we ought to devote more attention to the fact that the operators' conceptions of their core task and the actual competencies are formed in a societally mediated cultural environment. When putting the development of professional competencies into the cultural context we can reveal how the economical, safety and environmental goals become effective in actual operational decisions on the different levels of the organisation. As a consequence, construction of safety culture in practice will be included in our future studies as a necessary extension of our analysis of the working practices in the NPP operations.

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Development of contextual task analysis for NPP control room operators' work

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Abstract

The paper introduces a contextual approach to task analysis concerning control room operators' tasks and task conditions in nuclear power plants. The approach is based on the ecological concept of the situational appropriateness of activity. The task demands are dependent on the ultimate task of the operators which is to maintain the critical safety funcions of the process. The context also sets boundary conditions to the fullfilment of these demands. The conceptualisation of the context affords possibilities to comprehend and make visible the core demands of the operators' work. Characteristic to the approach is that the conceptualisation is made both from the point of the operators who are making interpretations of the situation and from the point of the process to be controlled. The context is described as a world of operators' possibilities and constraints and, at the same time, in relation to the demands set by the nature of the process.

The method is under development and has been applied in simulator training, in the evaluation of the control room information and in the integrated development of reliability analysis. The method emphasizes the role of explicit conceptualisation of the task situations. Explicity enhances its role as a conceptual tool and, therefore, promotes common awareness in these domains.

1. Introduction

Conceptualisation of decision making situations has been a fundamental part of the methodology of the man-machine psychology group, and the evaluation of the nuclear power plant operators' task performance has been made in relation to the process control context (e.g. Hukki & Norros 1993, Hukki & Norros 1998, Norros & Hukki submitted). Decision making has been considered by emphasizing the actors' subjective interpretations of the task situation. These interpretations, forming the basis of the analysis, concern the process, the task demands, and the situational possibilities to fullfil these demands (Hukki & Norros 1998). The operators' habits of action has been the main interest in the evaluation. They have been defined as the operators' ways of taking into account the possibilities and constraints concerning the use of the available resources of decision making. They are expected to contribute to the situational adaptability of the process control actions and, therefore, to the adequacy of the task performances. The adaptability of activity, that is, its situational appropriateness, is the foremost criterion in the evaluation of the habits of action. The results of the previous studies suggest that there are orientation based differences in the operators' habits of action (Norros & Hukki submitted, Hukki & Norros 1998). Corresponding results have been obtained from other complex environments (Klemola & Norros 1997).

The current goal of the group is to comprehend the core demands of the control room operators' work (see Norros in this volume). In order to be able to find out and define these demands the essential features of the context from the point of operators' decision making should be comprehended. The aim of the approach introduced here is to develop a method which would help to identify the critical features of the process control situations in a nuclear power plant from the operators' decision making point of view (Hukki in preparation).

2. The significance of the context

Contextual task analysis is needed because the appropriateness of the task performance always depends on the situation.The task demands depend on the context which, at the same time, sets boundary conditions to the fullfilment of the task demands.

Task analysis has been an essential part of human factors research and there have been several approaches to contextual task analysis in the field. The present view is close to but also deviates from both the naturalistic decision making approach and the ecological approach, both quite prominent nowadays (see Hukki and Norros 1998). Characteristic to the approach is that the conceptualisation is made from the point of view of both the operator who is making interpretations on the situation, and the process to be controlled.

The context affords possibilities to infer and to act but also sets boundary conditions to the operatos' decision making. Therefore it is described as a world of possibilities and constraints and, at the same time, in relation to the demands set by the nature of the process. The possibilities are afforded by the available resources of decision making and the constraints derive from the prevailing boundary conditions concerning their use. The operators take these boundary conditions into account according to how they comprehend the task demands of their work.

Descriptions of the context are conceptual tools which are used as a reference in the analysis of the operators habits of action. They help to evaluate how the operators take the situational conditions into account and, finally, to find out the cognitive demands of fulfilling the process control task. The criteria of the cognitive demands used so far which have been based on orientational aspects have been rather general (Hukki & Norros 1998). The development of a more detailed definition has been in progress in order to be able to find out the situationally most important demands, i.e. the core demands of process control (Norros & Hukki submitted, Hukki in preparation).The core demands can be used as the criteria of the evaluation and comparison of the operators' habits of action and, in addition, in the classification of the process control situations from the point of view of decision making.

3. Principles of conceptualisation

The basic concept in the description of the context is the situational appropriateness of activity (Hukki & Norros 1998). The main goal of the operators is to produce electricity safely. The concept of the critical safety functions of the process (e.g. Corcoran et al. 1981) is here used as the criterium of the general task demand of the operators (Hukki & Norros 1998). According to that, the ultimate task of the operators can be defined as the maintenance of the critical safety functions. The conceptualisation of the context is made in relation to this task by taking, at the same time, into account the prevailing boundary conditions restricting their use. This means that the situational boundary conditions of fulfilling the task demands, that is, the factual possibilities to make inferences and to operate, are described in relation to the ultimate task.

The operational alternatives are the practical resources for controlling the process. The available information is a resource for making inferences concerning the process and the operations. Also the communication and cooperation of the crew serve as resources of decision making but they are not discussed here further. The boundary conditions setting constraints to the control room operations are technical, economical, safety related, normative and also cultural by nature.

The critical information needed in the interpretation and handling of the situation is described according to its sources and diagnostic and operational informativeness, noticing the possible deficiencies of this information which may occur especially in disturbance situations. The appropriateness of each operational alternative is defined according to its availability and usability in relation to the main goal and the boundary conditions of the situation. Availability is here defined as the functional condition of the methods, and usability as the operability, capacity and long term impacts of the methods. The prevailing operating culture also contributes to the specific conditions of the process control situation.

The ways of describing the context conceptually are here called the reference models.With the help of them the conceptualisation can be made explicit in different ways. It is useful to describe the informativeness of the information and the appropriateness of the operational methods in tables and to draw also flow diagrams concerning the operational alternatives in relation to the ultimate process control task. These descriptions serve as conceptual tools and it is important to notify that they are tentative by nature (see chapter 4). The construction of concrete descriptions is necessary before the operators' interviews or before the observation and evaluation of their task performances because they help to direct attention to the relevant features of the task situation in order to define criteria for the habits of action. They also help to make the operators' tacit knowledge visible in the interviews. In the case of nuclear power the task situation can be described rather comprehensively beforehand but in those domains where the process is not technically as well known it may be necessary to approach it on a more general level.

4. Benefits of conceptual tools

The conceptual descriptions based on the present approach have been developed for simulator training, for the evaluation of control room information and for an integrated approach to reliability analysis. Cooperation with the field experts is an important aspect in the construction of the descriptions. The benefits of the contextual analysis in these domains is shortly discussed in the following.

In simulator training the conceptual decriptions are used as a reference for the evaluation of the crews’ task performances. The use of the descriptions ties the training period into a coherent and systematic entity . Before the simulation run the trainers conceptualise the scenario by creating the descriptions. In the next phase, during the simulator run, the descriptions are used as tools for the observation and evaluation of the operators’ task performance and habits of action. During debriefing they are used as tools for giving and getting feedback and they serve also the basis for the discussion of the learning content of the scenario. After debriefing they are used for documentation and before the next simulator run for the design of the training scenarios.

The explicit conceptual descriptions help to make the situation-specific demands and boundary conditions visible. The operators become more aware of the nature of the disturbance situation and of the effects of their actions on the process. At the same time, the criteria of the adequacy of the task performance and the appropriate habits of action as prerequisites of adequacy become more explicit and systematic. As a consequence, not only the trainees’ but also the trainers’ basis of inference become more easy to discuss and evaluate. It is emphasized that the descriptions are preliminary and tentative because they are developing during the process of the analysis, according to the knowledge concerning the crews' real activity, and because they may change as a result of the discussion in the debriefing phase.

Conceptual tools based on the principles introduced here are included in the evaluation method which has been developed for the simulator training of TVO Olkiluoto nuclear power plant.

In the evaluation of the appropriateness of the control room automation and information systems the conceptual descriptions help to comprehend the operators' task demands and the ability of the systems to make it possible to fullfil these demands. The quality of the systems sets boundary conditions to the operators' possibilities to infer and to operate. By conceptualising the features of the different sources of information in relation to the operators' comprehensions of the demands set by different process control situations the significance of these features can be inferred. A study concerning the appropriateness of an alarm system in a NPP control room and based on interviews of the operators is in progress (Hukki in preparation). In addition to studies based on interviews the appropriateness of the control room systems can be evaluated by observing and evaluating operators' activities on-line and by using the habits of action as a tool of analysis (Norros, Holmberg, Hukki, & Nuutinen 1997 unpublished, see Norros in this volume).

The conceptualisation of the functions of the systems from the point of the task demands enhances the designers' and the users' common awareness of the users' tasks and task conditions and helps to develop the design criteria of control room information.

In the development of reliability analysis the group has been developing an integrated reliability analysis together with reliability engineers (see Holmberg, Hukki, Norros, Pulkkinen & Pyy accepted). The aim is the integration between probabilistic and psychological approaches in human reliability by creating a conceptual interface between the probabilistic reliability analysis and contextual psychology in order to make it visible how operators’ different ways of decision making contribute to safety. The starting point has been the analysis of a context by using a common set of conceptual tools. The descriptions have served as mediators between approaches which are based on different perspectives on human activity.

5. Discussion

The introduced way of constructing conceptual descriptions and using them as a reference in the evaluation of the operators' task performances and task conditions widens the perspective to task analysis. The present approach is based on an ecological principle of the situational appropriateness of activity. It is essential that the “task” is defined on a sufficiently global level, in relation to the most important demands set by the nature of the process to be controlled. The different ways of realising this ultimate, global task, are constructed during the course of action and manifest the operators’ habits of action.

Conceptualisation of the context can be more or less explicit, depending on the situation. Explicity increases the benefits of the conceptual tools by enhancing common awareness between different parties in the development of the safety of the nuclear power plants, i.e. between trainers and trainees in simulator training, between designers and users of control room systems and between psychologists and reliability engineers.

The fulfillment of the task demands is dependent not only on the operators' comptetence but also on the features of the context. The conceptual descriptions help to gain knowledge of the demands of different types of task situations. On the basis of that knowledge also the cognitive aspects of the process control situations can be taken into account in the design of the scenarios of simulator training and in the development of the design criteria of control room information (cf. Hammond 1993).

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Norros, L., Holmberg, J., Hukki, K. & Nuutinen, M. (1997). Validation of NPP control room information aid. Unpublished report.

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Appendix A: International co-operation

In the following catalogue the most important current international cooperation contacts for the RETU 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),* R. Kyrki-Rajamäki, 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, P. Pyy, 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, P.Pyy / 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,* T. Tommila, P. Pyy, B. Wahlström, VTT Automation

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

* *Effect of communication to team reliability,* J. Kettunen, P. Pyy, 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

* *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 - ISANEW, P.Pyy

Concerted Action on Organisational Factors - ORFA, B. Wahlström

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

* *Working group on VVER Reactor Safety Analysis*, R. Kyrki-Rajamäki, VTT Energy
* *Working group on physical problems of spent fuel, radioactive waste and decommissioning of nuclear power plants,* A. Tanskanen, VTT Energy
* *AER annual symposiums*, R. Kyrki-Rajamäki, VTT Energy

Nordic Nuclear Safety Research (NKS),

Steering group, L. Mattila, VTT Energy

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

Cooperation with various institutes

Kurchatov Institute, Moscow Russia

* *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

Science and Research Technological Institute (NITI), Sosnovy Bor, Russia

* *Investigation of decay heat removal processes to emergency and fuel pools in VVER-640 reactors with the PACTEL test facility*, H. Purhonen, 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

* *Human reliability in fire scenarios*, P.Pyy, VTT Automation

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

* *Co-operative PRA search Programme (COOPRA),* P.Pyy, VTT Automation (observer)

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

JRC Ispra

* *Expert judgement,* 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

European Association of Cognitive Ergonomics (EACE), L. Norros, VTT Automation

New Technology and Work (NeTWork), L. Norros, VTT Automation

Work process knowledge in technological and organizational development (WHOLE), Thematic network, TSER, L. Norros VTT Automation

Appendix B: Publications in the projects of the RETU programme in 1995 - 1998

Table of publications in the research fields or projects of the RETU programme in 1995 -October 1998.

Research field or project	Acronyms of the projects	Ref. to articles of this report	Scientific journals and books	Conference papers	Research institute reports	Others	Total
LWR fuel performance	PATRA & KOTRA	(2)		7		14	21
Reactor physics & dynamics	DYNAMIC	(3, 4, 5)	2	58	3	69	132
Thermal-hydraulic experiments and analyses for VVER and ALWR plants	TEKOJA & PAHKO	(6, 7)	3	18	2	39	62
Severe accident management	VAHTI & ROIMA	(8, 9)	3	40	10	39	92
Risk and reliability analyses	LURI	(10, 11)	3	28	15	13	59
Human factors in NPP operations	ORINT	(12, 13)	6	7	1		14
Total			17	158	31	174	380

Transient Models of Nuclear Fuel (PATRA), VVER-Fuel Experiments (SOFIT) and Transient Behaviour of High Burnup Fuel (KOTRA)

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.1997. 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.

Ranta-Puska, K., Fission Gas Release at Burnups from 50 to 90 MWd/kgUO2: ENIGMA Calculations against Data from IFA-597.2/3 and IFA-562.2-6. Paper presented at the Enlarged Halden Programme Group Meeting on High Burn-up Fuel Performance,Safety and Reliability. Lillehammer, Norway, 15 to 20 March 1998. 6 p., 14 figs.

Pihlatie, M., Ranta-Puska, K., Method for Probabilistic Fuel Behaviour Assessment Applied to Loviisa NPP. Paper presented at the Enlarged Halden Programme Group Meeting on High Burn-up Fuel Performance,Safety and Reliability. Lillehammer, Norway, 15 to 20 March 1998. 9 p., 13 figs.

Contribution of S. Kelppe to the report: Fuel Modelling at Extended Burnup. Report of the Co-ordinated Research Programme on Fuel Modelling at Extended Burnup - FUMEX 1993-1996. IAEA-TECDOC-998. International Atomic Energy Agency, Vienna 1998.

Others

Stengård, J-O.1995. 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.1996. Progress of development, validation and application of the SCANAIR code. Espoo: VTT Energy. PATRA-6/96.

Ikonen, K., Kelppe, S., Ranta-Puska, K., Mechanical Analysis of a Nuclear Fuel Rod in a Fast Thermal Transient, PATRA-3/97, 29.5.1997.

Kelppe, S., Aspects of Clad Ballooning Modelling in a Reactivity Transient, PATRA-4/97, 2.6.1997.

Kelppe, S., Development, Validation and Application of the SCANAIR Code, PATRA-5/97, 3.6.1997.

Ranta-Puska, K., Polttoainetutkimuksen uusi projekti vuosille 1998-2000: Korkeapalamaisen polttoaineen transienttikäyttäytyminen, PATRA-7/97, 26.9.1997.

Ranta-Puska, K., Validation for the ENIGMA Fuel Performance Code, Nuclear Newsletter, VTT Energy, March 1997

Ranta-Puska, K., Fission Gas Release at Burnups from 50 to 90 MWd/kgUO_2: Enigma Calculations against Data from IFA-597.2/3 and IFA-562.2-6. PATRA-9/97, 14.1.1998.

Ranta-Puska, K., Specific Heat of UO2 and MOX Fuel: A Literature Review. Technical Report KOTRA-1/98, VTT Energy, 18.6.1998. 17 p., 13 figs, app.

Reactor Physics and Dynamics 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.

Puska, E.K., Norrman, S., Kyrki-Rajamäki, R., Siltanen, P., Porkholm, K., Latyeva, L, Semishin, V. & Kalugin, S., Advanced process simulation with APROS for VVER-91, the next generation reactor design for VVER-1000. Accepted for publication in Nuclear Safety.

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Human Factors in NPP Operation (ORINT)

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Norros, L. and Hukki, K. Dynamic of process operators’ decision making in a disturbance situation. A contextual analysis. Accepted for publication in International Journal of Cognitive Ergonomics.

Norros, L. & Klemola, U-M. Methodological considerations in analysing anaesthetists’ habits of action in clinical situations. Accepted for publication in Ergonomics. (Funded from another project).

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.)

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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.

Hukki, K. & Norros, L. 1997. A systemic approach to the modeling of decision making in the control of complex dynamic processes. Proceedings of the 13th Triennial Congress of the International Ergonomic Association, June 29-July 4, Tampere, Finland, Volume 3,43- 45.

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Nuutinen, M. & Norros, L. 1998. Development of personal competence as prerequisite for safety and quality. Paper presented at the European WHOLE-network meeting in Leuven, June 15-17, 1998.

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.)

Appendix C: Academic degrees

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.)

Narumo, T. 1997. Modelling of two-phase flow based on separation of the flow according to velocity. Espoo. VTT Publications 319. 39 p. + app. 103 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.)

Vihavainen, J. 1998. Computer code analyses on gravity driven core cooling system for nuclear power plants. (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.)

In addition to these degrees, one German and two Italian students have passed their Master of Science examinations based on the PACTEL test facility results at Lappeenranta.

Appendix D: Steering group, reference groups and scientific staff of the RETU programme

Steering group of the RETU research programme

DTech Lasse Reiman, Chairman	Radiation and Nuclear Safety Authority (STUK)
MScTech Markku Friberg	Teollisuuden Voima Ltd
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 Ltd
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	Radiation and Nuclear Safety Authority (STUK)
MScTech Matti Ojanen	STUK
MScTech Pertti Siltanen	IVO Power Engineering Ltd
MScTech Martti Antila	IVO Power Engineering Ltd
MScTech Risto Teräsvirta	IVO Power Engineering Ltd
MScTech Seppo Koski	Teollisuuden Voima Ltd
MScTech Esa Mannola	Teollisuuden Voima Ltd
LicTech Ralf Lunabba	Teollisuuden Voima Ltd
MSc Lena Hansson-Lyyra	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 Ltd
DTech Juhani Hyvärinen	Radiation and Nuclear Safety Authority (STUK)
MScTech Samuli Savolainen	Imatran Voima Ltd
MScTech Markku Friberg	Teollisuuden Voima Ltd
DTech Markku Rajamäki	VTT Energy
LicTech Olli Tiihonen	VTT Energy
DTech Heli Talja	VTT Manufacturing Technology
MScTech Virpi Kouhia	Lappeenranta University of Technology

Reference group for the Reactor Accident Phenomena and Modelling (ROIMA) project

MScTech Kalevi Haule, Chairman	Radiation and Nuclear Safety Authority(STUK)
DTech Juhani Hyvärinen	STUK
DTech Harri Tuomisto	IVO Power Engineering Ltd
MScTech Olli Kymäläinen	Imatran Voima Ltd
MScTech Heikki Sjövall	Teollisuuden Voima Ltd
MScTech Seppo Koski	Teollisuuden Voima Ltd
MScTech Pertti Auerkari	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	Radiation and Nuclear Safety Authority (STUK)
MScTech Vesa Ruuska	STUK
DTech Jussi Vaurio	Imatran Voima Ltd
LicTech Kalle Jänkälä	Imatran Voima Ltd
MScTech Risto Himanen	Teollisuuden Voima Ltd
Mr Markku Malinen	Teollisuuden Voima Ltd
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) and Transient Behaviour of High Burnup Fuel (KOTRA)

Person	Tasks
MScTech Seppo Kelppe	Project manager, 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); validation of TRAB-3D
MScTech Markku Anttila	Reactor Physics; OECD/NEA connections; NEANSC (Science Committee)
MScTech Antti Daavittila	Reactor Dynamics: hot channel modelling, validation of TRAB-3D
Ms Milja Eskola	Application of PLIM in the reactor dynamics codes
MScTech Anitta Hämäläinen	Circuit modelling, dynamics benchmarks
LicTech Randolph Höglund	Reactor Physics, Nordic connections

MScTech Elja Kaloinen	Development and validation of TRAB-3D
DTech Riitta Kyrki-Rajamäki	Development and testing of HEXTRAN-PLIM (including circuit model) and TRAB-3D; International co-operation on VVER safety, eg. VVER dynamics benchmarks
Mr Saku Latokartano	Reactor Physics; development of the CORFU code (MScTech thesis in 1999)
Mr Riku Mattila	Reactor Physics; pin power reconstruction model (MScTech thesis in 1998-1999)
DTech Timo Narumo	Development and validation of SFAV (new 6-equation thermal hydraulics model); resigned from VTT in April 1998.
DTech Markku Rajamäki	Development, testing and application of CFDPLIM and SFAV (thermal hydraulic models and accurate solution methods)
MScTech Aapo Tanskanen	Reactor Physics; validation of criticality safety methods
Mr Miikka Taponen (part time)	Studies on advanced thermal hydraulics methods
DTech Timo Vanttola	Special questions on thermal hydraulics
LicTech Frej Wasastjerna	Reactor Physics; RBMK, MCNP (Monte Carlo-calculation)

The project also employed research trainees.

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
LicTech 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 the measurement qualities of the CAWP-method, habit of action concept, co-operative practices as a component of habti ofa cation, integrated safety analysis
MA Kristiina Hukki	Development of the taxonomy of process situations as a component of the CAWP-method (PhD work), development of information presentation
MA Maaria Nuutinen	Analysis of the process operators' coping with problem sitautions as a component of habit of action (PhD work), analysis of the measurement qualities of the CAWP-method

Appendix E: Follow-on research programme on nuclear power plant safety 1999 - 2002

The Ministry of Trade and Industry (KTM) has decided to continue the national research efforts on reactor safety in a single research programme after completion of the research programmes on Reactor Safety (RETU, 1995-1998) and Structural Integrity of Nuclear Power Plants (RATU2, 1995-1998). The national advisory group, commissioned by KTM, made a general plan for the new programme and for its organisation (KTM Studies and Reports 15/1998, in Finnish), where the recommendations of the international evaluation of the RATU2 and RETU programmes (KTM Studies and Reports 8/1998) and opinions of national expert panels were taken into account. The new programme, *The Finnish Research Programme on Nuclear Power Plant Safety (1999-2002)*, may be divided into three themes, that are **ageing**, **accidents** and **risks**.

The Technical Research Centre of Finland (VTT) coordinates the programme and also performs most of the research. The main funding sources are KTM, VTT, the Radiation and Nuclear Safety Authority (STUK), the Lappeenranta University of Technology and the nuclear power companies Imatran Voima Ltd (IVO) and Teollisuuden Voima Ltd (TVO). During the four-year period, the annual volume of the programme will be about 30 person years and the annual funding FIM 23-25 mill.

General objectives of the programme are to develop tools and practices for safety authorities and utilities, to support identification, assessment and implementation of safety improvements, to provide a basis for safety-related decisions, to educate new nuclear experts and to promote technology and information transfer.

The research field of **ageing** includes studies on phenomena limiting the applicability of mechanical and electrical components, fuel and constructions. The aim is also to model effects of water environment and irradiation exposure to the ageing phenomena and particularly effects of re-irradiation after annealing of pressure vessel material. In the field of structural integrity the main objective is to create and verify experimental and computational methods for assessing the remaining lifetime of components and their ability to withstand possible accident situations. Systematic methods to improve the reliability of non-destructive testing are developed and techniques for continuous monitoring are searched and evaluated.

The **accident** field covers fuel research, reactor physics and dynamics, experimental and calculational thermal-hydraulics and severe accidents. Transient and steady calculation tools are qualified for high burnup and for fuels used in the local reactors. In reactor physics the main objective is to ensure the reliable operation of the wide code system through validation and updating, and to educate new reactor physicists. In reactor dynamics the aim is to complement and validate the calculation system for complex reactivity accidents, such as ATWS, boron dilution and BWR stability, and to improve the thermal-hydraulic models and the solution methods of the dynamics codes. In the experimental thermal-hydraulics the aim is to expand the scope of VVER related integral tests further by utilising the PACTEL facility. The results are used for validation of thermal-hydraulic system codes. Separate effect tests on phenomena relating to e.g. thermal fatigue of the pipes are planned. In the field of severe accidents, various phenomena acute for the Finnish reactors are studied, such as response of pressure vessel lower head on corium, coolability of debris beds (possibly experiments), chemistry of fission product iodine and containment loads. Training tools for severe accident mitigation are developed.

The **risk** field covers topics on fire safety, programmable automation, methods of risk analysis and human factors. Fire safety studies include effects of smoke on modern high density electronics, numerical fire simulation, burning models, flame propagation, reliability of fire detection and extinguishing systems and combining fire risk with living PSA. Programmable automation research addresses questions of licensing of software-based systems for safety-critical applications. Risk informed decision making methods are promoted and licensing practices, as well as methods for risk importance and uncertainty are developed. Practical methods are constructed for the evaluation and development of working practices and safety culture in the plants.

Eleven research projects are started in the above research fields. The planned volume and funding of the programme in 1999 is summarised below.

Table 1. The projects of the new programme and their volume and funding in 1999.

	Acronym	Volume [person years]	Funding [FIM mill.]
Ageing phenomena	AGE	4.0	3.8
Structural integrity	STIN	2.7	2.7
In-service inspections and monitoring	INSMO	1.0	1.0
Behaviour of high burnup fuel in accidents	KOTO	1.3	1.7
Reactor physics and dynamics	READY	5.3	3.3
Thermal hydraulic experiments and code validation	TOKE	2.9	2.6
Modelling and simulant experiments of severe accident phenomena	MOSES	3.2	2.8
Fire safety research	FISRE	2.0	1.0
Programmable automation system safety integrity assessment	PASSI	2.1	1.2
Methods for risk analysis	METRI	3.0	1.6
Working practices in nuclear power plant operation	WOPS	2.1	1.3
Administration and information of the research programme	HALTI	0.8	0.7
Total		30.4	23.7