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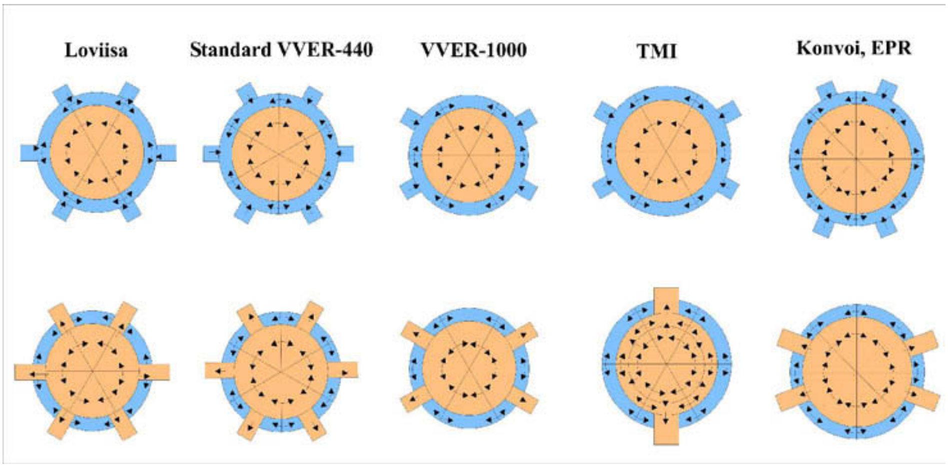

Anitta Hämäläinen

Applying thermal hydraulics modeling in coupled processes of nuclear power plants

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VTT Processes

Dissertation for the degree of Doctor of Technology to be presented with due permission of the Department of Energy and Environmental Technology at Lappeenranta University of Technology for public examination and debate in Auditorium 1381 on the 4th of November, 2005, at noon.

ISBN 951–38–6667–X (soft back ed.)
ISSN 1235–0621 (soft back ed.)

ISBN 951–38–6668–8 (URL: http://www.vtt.fi/inf/pdf/)
ISSN 1455–0849 (URL: http://www.vtt.fi/inf/pdf/)

JULKAISIJA – UTGIVARE – PUBLISHER

VTT, Vuorimiehentie 5, PL 2000, 02044 VTT
puh. vaihde 020 722 111, faksi 020 722 4374

VTT, Bergsmansvägen 5, PB 2000, 02044 VTT
tel. växel 020 722 111, fax 020 722 4374

VTT Technical Research Centre of Finland, Vuorimiehentie 5, P.O.Box 2000, FI–02044 VTT, Finland
phone internat. +358 20 722 111, fax +358 20 722 4374

VTT Prosessit, Lämpömiehenkuja 3 A, PL 1604, 02044 VTT
puh. vaihde 020 722 111, faksi 020 722 5000

VTT Processer, Värmemansgränden 3 A, PB 1604, 02044 VTT
tel. växel 020 722 111, fax 020 722 5000

VTT Processes, Lämpömiehenkuja 3 A, P.O.Box 1604, FI–02044 VTT, Finland
phone internat. + 358 20 722 111, fax + 358 20 722 5000

Technical editing Leena Ukskoski

Otamedia Oy, Espoo 2005

Hämäläinen, Anitta. Applying thermal hydraulics modeling in coupled processes of nuclear power plants. Espoo 2005. VTT Publications 578. 103 p. + app. 99 p.

Keywords thermal hydraulics, modeling, coupled processes, nuclear power plants, code validation, coupled codes, safety analysis, neutron kinetics, SMABRE, HEXTRAN

Abstract

This thesis focuses on the validation of the coupled codes developed in Finland for the safety analyses of the light water reactors in design basis accidents. The validation efforts and applications of the thermal hydraulics code SMABRE and the three-dimensional neutron kinetics codes are introduced for both the separate and coupled codes. The code development and coupling of codes for the safety analyses in Finland are discussed, and the present situation and possible future directions are described.

The data for the code validation consists of the experimental data from test facilities, numerical benchmarks and data measured in real nuclear power plants. The nuclear core is relevant for the couplings of neutron kinetics and thermal hydraulics codes. Two European Union projects, in which ten transients at real VVER plants have been documented, as well as the other international benchmarks are dealt with as useful forums in the code validation.

The simulation of plant measurements and several plant modeling aspects emerging from the validation work are gathered together. The main steam line break in different kinds of plants is dealt as an example of the application of the coupled codes in safety analysis. The coupling of a thermal hydraulic code with a fuel transient performance code is illustrated in the thesis as a new approach to performing the safety analyses. Another new approach is the sensitivity and uncertainty analyses performed for a Loviisa plant turbine trip case.

In addition to this summary, the thesis consists of seven publications in appendices: five articles in scientific journals and two conference papers.

Preface

The thesis has been prepared at VTT (Technical Research Centre of Finland) in connection with the activities of the Reactor Technology group. The validation work has mainly been funded by VTT, the National Research Programs and the European Commission. Several applications have been funded by Fortum Nuclear Services as well as by STUK, the Radiation and Nuclear Safety Authority in Finland.

I would like to thank my supervisor, Professor Riitta Kyrki-Rajamäki from Lappeenranta University of Technology, for encouraging me to prepare this thesis. She has taught me the reactor dynamics and, as a colleague at VTT, has made a valuable contribution to the success of many validation and application efforts.

Special thanks go to Mr. Thermal Hydraulics, Jaakko Miettinen, who has always guided me in thermal hydraulics problems; without him, the coupled codes in Finland and my whole career would have been totally different.

The development, validation and applications of the safety analysis codes have always been teamwork at VTT. I thank Dr. Timo Vanttola, Mr. Antti Daavittila, Mr. Elja Kaloinen, Mr. Jan-Olof Stengård, and all the others who have contributed to this work. The customer and co-author, Mr. Pertti Siltanen gets special thanks for his ideas and advice as well as Mr. Keijo Valtonen, who has promoted the latest couplings of codes.

As many of the authors of the articles in this thesis express, thanks also go abroad, to the colleagues in several countries.

Finally, I would like to thank my husband, Heikki, and my daughters for their patience during this effort.

Espoo, September 2005 Anitta Hämäläinen

List of publications

This thesis is based on the following seven publications:

I. Daavittila, A., Hämäläinen, A. & Kyrki-Rajamäki, R. Effects of secondary circuit modeling on results of PWR MSLB benchmark calculations with new coupled code TRAB-3D/SMABRE. Nuclear Technology, Vol. 142, No. 2, pp. 116–123, May 2003. [1]

II. Hämäläinen, A., Kyrki-Rajamäki, R., Mittag, S., Kliem, S., Weiss, F. P., Langenbuch, S., Danilin, S., Hadek, J. & Hegyi, G. Validation of coupled neutron kinetic / thermal-hydraulic codes Part 2: Analysis of a VVER-440 transient (Loviisa-1). Annals of Nuclear Energy, 2002. Vol. 29, pp. 255–269. [2]

III. Mittag, S., Kliem, S., Weiss, F. P., Kyrki-Rajamäki, R., Hämäläinen, A., Langenbuch, S., Danilin, S., Hadek, J., Hegyi, G., Kuchin, A. & Panayotov, D. Validation of coupled neutron kinetic / thermal-hydraulic codes Part 1: Analysis of a VVER-1000 transient (Balakovo-4). Annals of Nuclear Energy, 2001. Vol. 28, pp. 857–873. [3]

IV. Vanttola, T., Hämäläinen, A., Kliem, S., Kozmenkov, Y., Weiss, F.-P., Keresztúri, A., Hádek, J., Strmensky, C., Stefanova, S., Kuchin, A., Hlbocky, P., Siko, D. & Danilin, S. Validation of coupled codes using VVER plant measurements. Nuclear Engineering and Design, 2005. Vol. 235:2–4, pp. 507–519. [4]

V. Hämäläinen, A., Vanttola T. & Siltanen, P. Advanced Analysis of Steam Line Break with the Codes HEXTRAN and SMABRE for Loviisa NPP. In: OECD/CSNI Workshop on Advanced Thermal-hydraulic and Neutronic Codes: Current and Future Applications. Barcelona, Spain, 10–13 April, 2000. Issy-les-Moulineaux: OECD Nuclear Energy Agency, 2001, pp. 325–336. (NEA/CSNI/R(2001)1.) Full paper review. [5]

VI. Valtonen, K., Hämäläinen, A. & Cunningham, M.E. FRAPTRAN Fuel Rod Code and its Coupled Transient Analysis with the GENFLO Thermal Hydraulic Code. Proceedings of Nuclear Safety Research Conference (NSRC).

Washington, D.C., USA, 22–24 October, 2001. Washington, D.C.: US Nuclear Regulatory Commission, 2002, pp. 381–395. (NUREG/CP-0176.) [6]

VII. Langenbuch, S., Krzykacz-Hausmann, B., Schmidt, K.-D., Hegyi, G., Keresztúri, A., Kliem, S., Hadek, J., Danilin, S., Nikonov, S., Kuchin, A., Khalimanchuk, V. & Hämäläinen, A. Comprehensive uncertainty and sensitivity analysis for coupled code calculation of VVER plant transients. Nuclear Engineering and Design, 2005. Vol. 235:2–4, pp. 521–540. [7]

The author has specialized in the thermal hydraulics modeling of various experimental facilities and nuclear power plants. She has not written the codes used in this thesis but has been closely involved in the development through validation efforts and applications of the codes for safety analyses, especially in the SMABRE development.

The first paper focuses on the code validation in an International benchmark in an anticipated transient, the steam line break at the Three Mile Island plant. The comparison between several coupled codes is realized here. The author is responsible for creating the plant model, performing analyses with several variations and writing everything concerning the thermal hydraulics code SMABRE in this paper.

The next three papers, II, III and IV, consist of the validation of the coupled codes against measured VVER plant data as parts of European Union projects. The author is responsible for creating the plant models, performing analyses and reporting everything concerning the thermal hydraulics code SMABRE in these three papers. Further, in Papers II and IV she has the main role in the specification of the problems for the participants, in creating a system for the comparison between the measurements and the calculated results from several countries, and the writing of the papers.

The following paper, V, presents a final use of the coupled code, the safety analysis in an anticipated transient – the steam line break in the Finnish Loviisa plant. The author is responsible for creating the plant model, performing analyses and writing everything concerning thermal hydraulics and SMABRE code in this paper. The author also dealt with the steam line break of VVERs in [8 and 9].

New approaches to safety analyses are dealt with in the next two papers. Paper VI presents results of a new type of coupling of codes, the Finnish thermal hydraulic code GENFLO and the American fuel rod code FRAPTRAN. The author has the main role in realizing the coupling of the codes and is responsible for all the calculations and issues concerning thermal hydraulic code GENFLO in the paper. She also dealt with the coupling of GENFLO and FRAPTRAN in [10, 11, 12 and 13].

The last paper VII represents future means in safety analyses: sensitivity and uncertainty analyses performed for the Loviisa and Balakovo transients described in papers II and III. In this part of the EU project in Finland, the author had the main role in deciding the uncertainty parameters, creating a system to use HEXTRAN-SMABRE for the first time in this kind of analysis, and performing the calculations and reporting them.

Contents

Appendices I, IV, V and VII of this publication are not included in the PDF version. Please order the printed version to get the complete publication (http://www.vtt.fi/inf/pdf/)

Acronyms

AER	Atomic Energy Research co-operation
ATWS	Anticipated Transient Without Scram
BE	Best Estimate
BM	Benchmark
BWR	Boiling Water Reactor
CFD	Calculational Fluid Dynamics
DBA	Design Basis Accident
DNB	Departure from Nucleate Boiling
EU	European Union
EPR	European Pressurized water Reactor
GRS	Gesellschaft für Anlagen- und Reactor Sicherheit
ISP	International Standard Problem of OECD/NEA
LOBI	Loop for Off-normal Behaviour Investigation (test facility)
LOCA	Loss Of Coolant Accident
LOFT	Loss Of Fluid Test (Test facility)
LWR	Light Water Reactor
MCP	Main Coolant Pump
MSLB	Main Steam Line Break
NPP	Nuclear Power Plant
OECD /NEA /CSNI	Organisation for Economic Co-operation and Development /Nuclear Energy Agency /Committee on the Safety of Nuclear Installations

OTSG	Once-Through Steam Generator
PACTEL	PArallel Channel TEst Loop (test facility)
PIPER-ONE	Italian BWR – related test facility
PMK-NVH	Hungarian test facility for VVER
PWR	Pressurized Water Reactor
RCP	Reactor Coolant Pump
REWET	Finnish test facility
ROSA	Rig Of Safety Assessment (test facility)
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SPES	Simulatore PWR per Esperienze di Sicurezza (Test facility)
SPND	Self-Power Neutron Detector
SUSA	Software System for Uncertainty and Sensitivity Analyses
TMI	Three Mile Island (Nuclear power plant)
VTT	Technical Research Centre of Finland
VVER	Russian Pressurized Water Type Reactor
YVL	Regulatory Guides on nuclear safety in Finland

1. Introduction

In nuclear power plants, taking account of possible incidents is essential for safety and economical reasons. In this effort the tests at real power plants include only a small part of the anticipated incidents; the majority of incidents, and accidents in particular, have to be simulated with computer programs. The simulation programs should, however, be validated. Here, the earlier documented incidents as well as the experiments at real plants and test facilities have an important role.

The various physical processes in a nuclear reactor include coupled physical phenomena with strong interactions. In spite of the increasing computing capacity, all these processes cannot be modeled in one single program. Special separate programs and effective couplings between them are needed. This thesis focuses on the validation of these coupled codes, which the combination of independent codes calls for. The thesis gathers up the validation efforts for the codes involved, and outlines the present situation and trends in the coupling of codes in advanced analyses of the safety-related issues for nuclear reactors. Some EU projects and international benchmarks are considered a useful forum in the code validation in several types of transients in several different kinds of PWRs. The final use and target of the developed codes is demonstrated via one application in safety analysis, the Main Steam Line Break (MSLB) in the Loviisa plant.

The code development in nuclear safety analyses at VTT, The Technical Research Centre of Finland, is discussed in Chapter 2, where the present situation and future directions are described. Generally, the dissertation focuses on the Finnish efforts in the coupling of thermal hydraulics and neutron kinetics codes in the transients belonging to the design basis accidents (DBA), especially in the transients where three-dimensional (3-D) neutron kinetics is needed. In this field the codes have traditionally been developed in separated parts: development of static neutronics and reactor dynamics codes, development of thermal hydraulic system codes and development of fuel rod codes. The present target for safety analyses is to create a compact data exchange between codes and to couple all the necessary parts in one calculation system. A common feature in the development of all codes is that the calculation mesh gets more detailed all the time. Still, the codes should have a reasonable consumption of computing time with such a detailed system.

Typically, the first tool for performing safety analysis on the plant scale is a thermal hydraulics system code. Heat and flow sources and leaks belong to the main boundaries in the calculation. For several purposes and transients, a simple decay power curve is sufficient to define the power generation in the reactor core. System codes often include point kinetics to improve the simulation of the core behavior. The point kinetics solution corresponds to 0-D neutronics and neither the radial nor the axial power distribution can change during a transient.

As a further improvement, the coupling of 1-D and then 3-D neutron kinetics codes to thermal hydraulics codes enables the analysis of ATWS and reactivity initiated accidents (RIA) with a core asymmetry, and illustrates more precise core behavior. However, thermal hydraulics is constantly calculated one-dimensionally. Some 3-D elements have been introduced in the 1-D calculation by modeling several parallel channels in the core. The lack of a usable 3-D thermal hydraulic model is obvious. Here the Computational Fluid Dynamics (CFD) codes give an opportunity in this field even if the available two-phase CFD-tools are far less mature than the one-phase tools [14]. In practical safety analyses, the appearance of steam as the second phase greatly increases the computing time with these codes.

There are several models for extending thermal hydraulics system code for 3-D, such as the American RELAP5-3D [15], the European SRELAP-5 modified by Siemens Power Corporation [16] or the special models of TRAC [17] and CATHARE [18]. Here the starting point has mainly been 2-D or 3-D components, not a general solution. For the core, 3-D thermal hydraulics has been introduced with, e.g., a traditional subchannel code, COBRA [19], in various versions, or the FLICA-4 [20] code, by coupling them to a neutron kinetics code and a system code. 3-D effects have also been described with the present 1-D system codes by a dense nodalization [21]. One solution for general 3-D thermal hydraulics could be the porosity model, which is under development at VTT [22]. In Finland, the actuality of 3-D thermal hydraulics simulation in the core is emphasized by the fifth nuclear power plant, Olkiluoto 3, which will be of the EPR type. EPR is a typical PWR plant including an open core without shrouds around the fuel assemblies, leading to a stronger mixing in the core. The other four operating reactors in Finland have shrouds around the fuel assemblies. Each assembly defines a flow channel, where thermal hydraulics is solved one-dimensionally.

With the fuel rod codes, the behavior of the fuel rods is normally analyzed in detail. For a rod, detailed axial and radial distributions of materials and burnups are given and the radial and axial geometry changes in the fuel and the gas gap due to mechanical and chemical reactions are simulated. These codes are typically used in the final stage to find the integrity of the fuel rods. Feedbacks to surroundings to define the conditions of the fuel rod's outer surface are not usually available, and, likewise, neutron kinetics is not used to define the power generation in the rod – it has to be presumed.

Chapters 3 and 4 in this thesis focus on the coupling of neutron kinetics codes with thermal hydraulics. The main validation efforts for the separated and coupled codes involved are summarized. Although, due to their non-nuclear cores, the integral test facilities only give limited data for the coupled code validation, there are several possible approaches, such as the validation against data from experiments on test reactors, start-up tests and real transients at real power plants, and hypothetical transients used in code-to-code validation.

The international benchmarks of OECD and the benchmarks of AER have mainly been numerical benchmarks, code-to-code validations. These frameworks for code validation slightly differ from each other. Typically, the specifications of the OECD benchmarks have been more precise in giving plant data – e.g., nuclear data – and emphasizing the comparison of codes. In the AER benchmarks only the institutes having their own VVER-models have participated and the benchmark has been closer to a rehearsal of safety analysis.

A valuable addition has been the European Union projects, where several transients at real VVER plants have been documented and made available for all participants for validation purposes. In the PHARE project SRR1/95 [23, 24], five transients were documented, three measured at VVER-1000 and two at VVER-440. For both VVER types, one transient was chosen for the code validation. The results are reported in Papers II and III, and the HEXTRAN-SMABRE calculations are referred to in Chapter 4.

In the EU/VALCO project [25], the data collection was extended to new types of transients and the validation of coupled codes was continued (Paper IV). Here too a collection of five transients was made, three concerning a VVER-440 and two a VVER-1000 reactor. Two of the new transients were chosen for the code validation.

Chapter 5 focuses on coupled code applications in safety analyses. The simulation of measurements and several aspects emerging in the validation work are gathered together. Furthermore, the main steam line break analysis in the Loviisa plant is presented as an example of the application of coupled codes in safety analysis.

Chapters 6 and 7 conclude the present trends in safety analyses. First, the coupling of a thermal hydraulics code with a fuel rod code is introduced (Paper VI) as an improvement to the last phase in the safety analysis. Second, for performing the safety analyses of actual transients, the method of uncertainty and sensitivity analysis is introduced. The method backs up the best estimate (BE) safety analyses because the conservatism of all assumptions can hardly be guaranteed. In the VALCO project sensitivity and uncertainty analyses were performed in several countries (Paper VII) for two transients, the other being the single turbine trip in Loviisa already used for validation in a previous EU project.

The main results are summarized in the conclusions, and the importance of code validation against plant data and the need for 3-D thermal hydraulics are emphasized.

This thesis is based on the seven Papers, mainly dealing with coupled code validation. The research work has been carried out at VTT within different research teams. In order to give readers the frames and background to the author's work, the VTT working tools, codes, validations and applications are briefly described here, even though not all the work has been carried out by the author personally. However, the role of each topic as part of this dissertation, as well as the author's contribution to these issues, are given in the beginning of the chapters.

2. Code development

The working tools, the codes and the couplings used for the calculations in the papers of this thesis are described in this chapter. Further, the present situation and future trends in code development are dealt with. It is underlined that the author has not written the codes herself.

The deterministic transient analyses of nuclear power plants are strongly based on the computer codes because experimental analyses with real materials or on a real scale are not easy to perform in such a complicated system. The experiments are often expensive, and sometimes impossible due to safety reasons. The used codes should, however, be validated and experiments have an important role here.

The most usual thermal hydraulic system codes for LWR transient analyses are the American RELAP5 [26] and TRAC, the German ATHLET [27] and the French CATHARE [28]. In Finland, the Finnish-made APROS [29] and SMABRE [30] are mostly used. It is possible to create several kinds of thermal hydraulic systems with these codes. In the safety analyses of DBA, typical flow media are water and steam – together forming a two-phase flow – mixed with non-condensable gases and boric acid. The solid structures and fuels may typically consist of several materials. The thermal hydraulic solutions in the system codes are quite similar, based on the conservation equations for mass, momentum and energy, but the number of conservation equations may vary. Due to simplifications and insufficient modeling, all phenomena cannot be modeled with these codes. The more complete methods, such as CFD, still need enormous computing time in transients with two-phase flow. Thus the codes are only relevant when calculating some special phenomena, such as flow mixing in a system.

One undesirable phenomenon is the numerical diffusion appearing in the solutions to the system codes, leading to an inability to simulate the progress of boron and/or temperature fronts in fluids. One solution could be the method of characteristics, such as in the CFDPLIM code developed at VTT [31], but until now it has not been applied in system codes due to the large computing capacity requirement. PLIM, the Piecewise Linear Interpolation Method, represents a totally different, accurate solution method for thermal hydraulics. It has been included in the core channel models of HEXTRAN and TRAB, and used in the boron dilution benchmark including boiling [32].

The present codes of VTT – categorized according to their main issue – the development going on and the future challenges are shown in Table 1. One possible future improvement in thermal hydraulics may be based on porosity models. At VTT the first steps have been taken in developing a 3-D thermal hydraulic code, PORFLO [22], which is suitable for several purposes, and because of its ability to vary node volumes in a modeled system. Here a coarse nodalization of large one-phase tanks modeled in the same system with small fluid nodes needed to be calculated 3-dimensionally inside a fuel assembly because of local effects. In this context, a fast iteration method is needed.

An OECD standard problem has been started [33], where code capabilities are compared against data measured in a full-size BWR bundle. The dynamically measured void mesh is radially dense, 0.3 mm x 0.3 mm, giving excellent opportunities for testing code capabilities. The detailed simulation of thermal hydraulics together with the pin-power reconstruction in neutron kinetics raises the core simulation in transients to a new level.

The starting point in transient analyses is the basic nuclear libraries used to create the cross sections for the 3-D core models. The CASMO-4 code [34] and its hexagonal version, CASMO-4E [35], are currently used for this purpose at VTT. In Finland, the nodal distributions for fuel burnup, void and control rod histories are obtained from fuel management codes, e.g. SIMULATE [36], or as a new option, ARES [37], for the square lattice geometry, and HEXBU-3D [38] for the hexagonal lattice geometries.

The created cross sections and burnup distributions are needed for dynamic core calculations, where cross sections during a transient are computed from polynomial fittings to fuel and coolant temperature, coolant density and soluble boron density. VTT's main tools for 3-D core simulation are the HEXTRAN [39] and the TRAB-3D [40] codes, either as stand-alone or coupled to SMABRE. Both the HEXTRAN and the TRAB-3D codes include sophisticated 3-D nodal core models. The codes are best-estimate types, but include special modeling features for taking account of conservative assumptions. The transient is calculated with HEXTRAN-SMABRE when the core lattice is hexagonal, or with TRAB-3D-SMABRE when a square core lattice is involved. Typically in transient analyses, each assembly creates one flow channel, where thermal hydraulics is solved one-dimensionally. An assembly is a natural choice for a channel even if there are no shrouds around assemblies because of the almost equal radial fuel burnups resulting from assembly-wise fuel re-loadings. Further, a natural choice for dividing the assembly into smaller parts in BWRs is, e.g., the four channels/subchannels formed by the water cross.

The need for a 3-D neutronics simulation is most pronounced in cases with asymmetric behavior of the reactor core and in cases without proper operation of the reactor trip (ATWS cases). The reasons for asymmetric core behavior may be outside the core, such as steam line breaks in PWRs and steam line isolation valve closures or turbine trips in BWRs. Asymmetric boric acid concentration in the core inlet is a largely analyzed VVER transient, which may occur after the start-up of an inactive loop. Asymmetric control rod movements in the core naturally lead to asymmetry. Further, oscillation transients generated in the core cannot be simulated without 3-D neutronics. In order to reduce extra conservatism, the large break LOCA [41] is best analyzed with 3-D neutron kinetics/thermal hydraulics codes. In Finland, the main coolant pump trips, startups and seizures have also been calculated with 3-D neutronics. An advantage of 3-D analysis in the core is the ability to model heterogeneous cores with mixed loadings of different fuel assembly types. These, currently typical, BWR cores outline the need for a denser nodalization or pin-power reconstruction in nodal dynamic codes.

The present targets in the development of reactor dynamics codes in Finland are dealing with a better modeling of heat transfer in the fuel rod, such as the modeling of dynamic gas gaps, and the modeling of rod internal pressure as a function of burnup. The importance of these models is decisive if high burnup fuel is used.

In order to find the final safety margins for the fuel, e.g. DNBR (departure from nucleate boiling ratio), and the maximum fuel rod enthalpies and cladding temperatures, hot channel calculations are performed. The dynamic boundary conditions of the core, represented by an average rod in the assembly, created with coupled codes, are afterwards used in these hot channel analyses. Typically, the hot channel coefficients are varied and several hot channel calculations are performed at VTT with the 1-D neutron dynamic code TRAB [42, 43].

Table 1. VTT's code system categories according to their main issue.

<table>
<tr><td></td><td>Thermal hydraulics</td><td>Neutronics</td><td>Cross sections</td><td>Fuel behavior</td></tr>
<tr><td rowspan="2">Static codes</td><td rowspan="2"></td><td>HEXBU,
SIMULATE3,
ARES</td><td>CASMO-4,
CASMO-4E</td><td rowspan="2">FRAPCON,
ENIGMA</td></tr>
<tr><td colspan="2">Monte Carlo codes</td></tr>
<tr><td rowspan="2">Present
dynamic codes</td><td>SMABRE,
GENFLO,
COBRA</td><td>TRAB,
HEXTRAN,
TRAB-3D</td><td rowspan="2"></td><td rowspan="2">FRAPTRAN,
SCANAIR</td></tr>
<tr><td colspan="2">APROS</td></tr>
<tr><td rowspan="2">Phenomena
under testing/
development
and future
needs,
and the
corresponding
codes</td><td rowspan="2">Micro and macro
geometry,
PORFLO

3-D thermal
hydraulics,
PORFLO,
CFD-codes

Elimination of
numerical diffusion,
preserve fronts,
CFDPLIM

Varying fluid
volumes, GENFLO,
severe accident
codes</td><td>Dynamic gas
gap,

High burnup,

Rod failure,

Xe-dynamics</td><td>Wide range
cross sections,</td><td rowspan="2">High burnup,

Pellet-cladding
(mechanical)
interactions
(PCMI, PCI)</td></tr>
<tr><td colspan="2">Pin-power reconstruction,
dense nodalization</td></tr>
<tr><td>New demands
in future plants</td><td colspan="4">Super-critical circumstances, various kinds of geometries in core, various fuel and cooling materials</td></tr>
</table>

On the other hand, the same boundary conditions from the transient calculations are used for the hot rod analyses with fuel transient performance codes to find the final fuel integrity in a transient. There are two calculation chains available for fuel behavior analyses in Finland. One is based on the British ENIGMA steady state code and the French transient code SCANAIR. The other combination is the steady state code FRAPCON-3 and the transient code FRAPTRAN, both from the U.S. Nuclear Regulatory Commission (USNRC). The steady state codes can be used separately or to provide a burnup-dependent initial state for a transient analysis.

The hot channel and hot rod analyses at VTT may be performed with a new coupled code, FRAPTRAN-GENFLO, allowing the transient fuel behavior analyses with a versatile hydraulics model. The code is described in Paper VI and Chapter 6. The need for 3-D thermal hydraulics is also obvious in hot channel analyses in order to avoid over-conservative results. The use of separated hot channels is even more questionable when a core without shrouds is concerned. The traditional subchannel code COBRA and its several modifications are still the few public codes able to simulate cross flows in hot channel analyses.

At VTT, the PWR and BWR calculation system is about the same. If, instead of the coupled codes, APROS with a 1-D neutron kinetics core model is used, the cross sections have to be created especially for 1-D. Typically with APROS, some hot channel calculations are performed at the same time as the transient calculation itself. APROS, like many simulators, differs somewhat from the coupled codes in transient analysis. For example, APROS has a graphical interface including special tools for the complete modeling of automation systems.

‘The complete code’ for safety analyses cannot be created. The phenomena involved are numerous and more and more features and details need to be modeled when the calculation mesh becomes denser. In Finland, with limited resources, the solution could be the continuation of couplings of all relevant codes and solution methods; at least, all the modules developed have to be compatible with the existing codes.

The fuel behavior and the simulation of severe accidents have generated needs for modeling chemistry and mechanical interactions in the codes. Further, the future plants will create new demands for updating the present codes.

The thermal hydraulic code SMABRE and the 3-D neutron dynamics codes HEXTRAN and TRAB-3D are briefly presented in the following chapters.

2.1 SMABRE model

The development of the SMABRE [30, 44, 45, 46] code started at the beginning of the 1980s. The development of the model originated from a practical need for a fast-running thermal hydraulic model for the scoping studies of small-break LOCA accidents. The SMABRE (=SMAll BREak) model is based on a non-iterative algorithm of five conservation equations, mass and energy for water and steam, and a single momentum equation for the mixture of steam and water. The phase separation is solved using the drift flux model.

The integral momentum solution of SMABRE combined with the system pressure concept was the starting point in the SMABRE development. In this computing time-saving solution, momentum equations are only solved at one point in each separate flow path. This solution is still an alternative solution to the full-momentum equation solution. The system pressures are applied to define some material properties in each partial system. These pressure systems are also used for simplifying the input/output operations – e.g., by defining a constant heat transfer coefficient for radiation to the surroundings. Otherwise, SMABRE is quite similar to the typical system codes having models for all the necessary components of power plants and general trip logic to create the automation.

The selection of constitutive models is presented in Table 2. The models take account of the phenomena during two-phase natural and forced circulation as well as during the blowdown phase of small and medium LOCAs. Simplified and modified correlations have been used for the heat transfer, but for typical LWR cases the differences from the original correlations are quite small – e.g., the interfacial heat transfer coefficients are smoother than in several system codes. Some differences in correlations to typical system codes exist due to the non-iterative solution of SMABRE, which can tolerate large time steps [30].

The point kinetic model simulates one energy group for neutrons and 6 precursor groups for the generation of delayed neutrons. The reactivity may be defined by simple reactor feedback coefficients or in a table form. The reactivity feedback is calculated as a function of the average liquid density and temperature, average fuel temperature, and boric acid concentration in the core.

Table 2. The main constitutive thermal hydraulic models for SMABRE [30].

Physical phenomena	SMABRE model
Wall friction	Blasius equation for mixture
Net vaporization	A linearized ramp function from subcooled liquid to saturation point
Pre-DNB heat transfer	Dittus-Boelter, Chen as simplified for boiling
Critical heat flux for wall heat transfer	Zuber-Griffith, VVER: Smolin, Bezrukow
Post-DNB wall heat transfer	Dittus-Boelter for gas
Interfacial condensation	Droplet type condensation or through stratified water level
Interfacial flashing	Linear function of liquid mass and liquid superheat
Critical flow limitation	Sound velocity limitation or Moody model applied for the junction
Pump characteristics	Four quadrant curves for head and torque for flow and pump speed.
Phase separation	Drift flux model derived from EPRI correlation or full separation
Material property solution	Rational function fittings, two- or one-parameter functions

The numerical solution for the SMABRE model is a predictor-corrector-type non-iterative solution. The sparse matrix inversion is used for solving the pressure, void fraction and enthalpy distributions. The pressure solution implicitly includes the result for the flow distribution. The use of the sparse matrix in the SMABRE solution has proved to be fast in the geometries modeled up to now, but in larger nets – e.g., in a dense nodalization for describing 3-D thermal hydraulic effects – a new solution method may be needed [30].

2.2 HEXTRAN and TRAB-3D models

The 3-D reactor dynamics code HEXTRAN [39] performs neutronic, fuel heat transfer and thermal hydraulic calculations within the reactor core. It solves the two-group neutron diffusion equations by the same nodal expansion method as the static core design code HEXBU-3D that is used in the fuel management calculations of the VVER reactors. VTT's other 3-D neutronics code, TRAB-3D [40], is based on one HEXTRAN version. The most significant modification here is that TRAB-3D uses square fuel assembly geometry.

The thermal hydraulic conservation equations are solved in separate parallel channels. Channel thermal hydraulics is based on four conservation equations for steam and water mass, total enthalpy and total momentum, and on a selection of correlations describing evaporation and condensation, slip, and one- and two-phase friction [42]. The phase velocities are related by an algebraic slip ratio or by the drift flux formalism.

A fuel temperature calculation is made for one average fuel rod in each fuel assembly. The fuel and cladding are discretized with several radial mesh points in the calculations, repeated at different axial elevations. The thermal properties of the fuel pellet, gas gap and fuel cladding are functions of local temperature and burnup, and the heat transfer coefficient from cladding to coolant depends on the thermal hydraulic regime. The fission power is divided into prompt and delayed power parts, and part of the power can be dissipated into heat directly in the coolant.

Advanced time integration methods are applied in the dynamic calculation. The numerical technique can vary between the standard fully-implicit theta method to the central-difference theta method, both in the heat conduction calculation for fuel rods and in the solution of thermal hydraulic conservation equations for cooling channels. The solution of equations is not based on nodes and junctions, but spatial and temporal discretization is used in the flow channels and the solution is optimized to stay near the characteristics [47].

TRAB-3D includes a BWR circuit model containing 1-D descriptions for the main circulation system inside the reactor vessel, including the steam dome with related systems, steam lines, recirculation pumps, incoming and outgoing flows, and control and protection systems [42].

Both HEXTRAN and TRAB-3D are best-estimate codes. The ability to modify the neutronics parameters has been included in the code so that the conservatism of the calculations can be simply and reliably modified without changing the ordinary neutronics data. These modifications to include conservative assumptions have been regularly applied in HEXTRAN analyses for VVERs.

The need for the HEXTRAN code may increase in future plants where the hexagonal core lattice is planned in some concepts.

2.3 Coupling of codes

The separate development of neutron kinetics and thermal hydraulic codes has led to the necessity to couple the codes as the transients and accidents to be analyzed have a strong coupling of neutronics, heat conduction and fluid dynamics. Furthermore, the analyses with stand-alone codes would presume time-dependent boundary conditions introducing unacceptable uncertainties.

The couplings of the thermal hydraulic codes and neutron kinetic codes have been a typical approach in reactor dynamics in OECD countries. The neutron kinetic codes are typically 3-D nodal codes and thermal hydraulic codes are large system codes. At VTT the coupling of codes [48] was started in the 1980s. The utilization of existing codes and models saves the limited resources of a small country and the results, as usable codes, may be achieved in a reasonable time.

Three types of coupling have been used in the coupling of thermal hydraulics codes and neutron kinetics codes: external, internal and parallel [Papers II–IV]. With the external coupling, all calculations in the core, including neutronics, thermal hydraulics and heat transfer, are performed with the neutron dynamic code. The data at the core inlet and outlet is then transferred to the thermal hydraulic system code for the calculation of the rest of the reactor circuit. When the internal coupling is used, the system code takes care of all the thermal hydraulics calculations both in the core and in the circuit. The core data is transferred between the codes node by node.

In the parallel coupling, the core thermal hydraulics is calculated by both codes. Thus the neutronics code takes care of the core calculations, including thermal hydraulics. Power distributions are transferred node by node to the system code, but the thermal hydraulics data is only transferred at the core inlet and outlet. In the parallel coupling applied to HEXTRAN-SMABRE, the two totally independent codes are coupled together. The thermal hydraulics of the core are calculated with both codes and the hydraulic equations around the whole circuit are only solved in the thermal hydraulic code. At VTT the first applications with coupled 3-D neutronics/thermal hydraulics codes were performed in 1991–1992 with HEXTRAN-SMABRE [49]. Earlier, the coupled 1-D neutronics/ thermal hydraulics code SMATRA was in use in 1988 [50]. The newest coupling, TRAB-3D – SMABRE, has been carried out in a similar way to the earlier ones.

With loose couplings, the numerics and solution methods of the codes may be different, as in the present HEXTRAN-SMABRE, iterative inside the core model and non-iterative in the circuit model. Separate development of the codes may be used in the coupled code. The modeling of the core may be different for these codes, such as a sparse nodalization of the core for SMABRE and a detailed nodalization for HEXTRAN and TRAB-3D.

In the parallel coupling the coupled code has its own main program and a few interfacing subprograms, but in the combination, HEXTRAN or TRAB-3D and SMABRE are used as if they were separate codes. One of the advantages of having separate codes is that the transients can be divided into three time periods. First, SMABRE alone calculates the stationary state at the initial power level. At the beginning of the transient HEXTRAN or TRAB-3D creates a compatible stationary state for the core and the two codes are coupled for the common calculation. During this period SMABRE typically consumes only some percentage of the calculation time. In very long transients after the main phase of a transient, when a 3-D core calculation is no more essential, the calculation may be continued with SMABRE alone.

Typically, the core modeled for SMABRE consists of as many parallel sectors as there are loops in the plant. The sectors are discussed in more detail in Chapter 5.2. In the HEXTRAN or TRAB-3D core model each fuel assembly is normally divided into 20 to 25 axial nodes for thermal hydraulics, neutronics and heat transfer. Typically, each assembly corresponds to one flow channel, but several assemblies can be combined into a flow channel too. Further, if core symmetries are applied instead of the whole core modeling, averaged parameter values of the SMABRE channels are used for HEXTRAN or TRAB-3D.

At present, the development of using the internal coupling between TRAB-3D and SMABRE is ongoing [51]. This will enable the modeling of the cross flows between channels with SMABRE in open cores as well as flow reversals in core channels (presently not possible). Furthermore, this kind of coupling may foretoken the possibilities of combining other thermal hydraulics and reactor kinetics codes together. However, the nodal neutron kinetics codes are very sensitive to thermal hydraulics and presume a more exact thermal hydraulic solution for the codes. The possible thermal hydraulics model could be APROS or the new porosity model, PORFLO, with 3-D thermal hydraulics. The alternative neutron kinetics code could be SIMULATE or ARES, further developed to include the time dependences of the neutron fluxes. The possible requirements for uncertainty analyses, introduced in Paper VII and Chapter 7, may be the reason against the coupling of several codes because of the need for reasonable computing time in uncertainty analyses. On the other hand, the transients may lead to totally different scenarios in the uncertainty analyses and may require more completed plant models, codes and couplings.

The coupling of a thermal hydraulic code with a fuel transient code and codes in the field of severe accidents is described in Chapter 6.

3. Code validation

Five of the papers in this thesis deal with coupled code validation. This chapter presents the validations of the codes involved. The author has participated in several ISPs and TMI exercise 1 with the SMABRE code, and has applied SMABRE in several applications. She did not take part in the reactor dynamics codes validation until the couplings of SMABRE had been realized with the 3-D codes. The author had the main role in applying the SMABRE system code in the three steam line break cases presented.

The safety analysis codes should be validated – that is, a representative set of calculations tested against measured or otherwise acceptable data. The validation is necessary to ensure the completeness and correctness of the codes. Thus the validation has an essential role in the code development, especially in Finland.

As an initiative of OECD/CSNI, numerous internationally agreed test facilities and tests have been gathered in test matrices to be utilized in the validation, assessment and improvement of best-estimate thermal hydraulic codes. The matrices are categorized as separate effect test matrices [52] or according to the reactor, steam generator and transient types [53]. Several Finnish experiments have been included in the matrices for VVERs [54]. Parallel to the advancement in code development, the experiments were initially focused on large break issues in the early 1970s. Nowadays integral tests have been carried out to investigate LWR system behavior – e.g., under shutdown conditions and beyond design basis accidents [55]. The storing of experimental data in the OECD/NEA databank is ongoing. The matrices create a systematic basis for the validation of many thermal hydraulic system codes. Typically, the ISPs are tests in the scaled down facilities and there may be uncertainties in scaling up predictions of phenomena from integral test facilities to real plant applications. That is why the counter-part tests – similar tests on a different scale – are considered highly important for code validation [55]. Another important group of tests are the tests behind the several international standard problems (ISP) organized by OECD. On the other hand, the international numerical benchmarks and tests on real plants are cases for the reactor dynamic code validation.

In the following, the system code validation is discussed and the SMABRE validation and application cases are gathered. Second, the cases used in reactor dynamic code validation, and, finally, the cases in the coupled code validation are gathered. The validation against real plant data is described in Chapter 4.

3.1 Validation and applications of SMABRE code

The first step in thermal hydraulic code validation is the validation against separate effect tests. In Finland, the starting point in the experimental field has been the tests dealing with VVER-440 reactors performed by both VTT, together with Lappeenranta University of Technology (LUT), and Fortum – at that time IVO. The IVO tests concentrated on the counter current flow limitation (CCFL) phenomena, thermal mixing and loop seal behavior in VVERs [56, 57]. In the VTT-LUT tests the rewetting phenomena in the overheated VVER core were measured in REWET tests and the results were used for the validation of several codes [58, 59, 60, 61, 62]. Later on, the PACTEL facility was built in Lappeenranta for integral tests, including one OECD Internal Standard Problem, ISP-33 [63, 64].

The experimental work in Lappeenranta has continued with several facilities and is not limited just to VVER. Generally, the understanding of the phenomena in transients for PWR has increased over the years, as well as the test facilities' abilities to demonstrate the phenomena.

The results from the tests performed in Finland have been used for the validation of the SMABRE code. The real plant startup tests and incidents in Loviisa have also been basic data for validation. With the stand-alone code SMABRE, VTT has participated in several ISPs organized by OECD and IAEA. Further, the SMABRE code is widely used and validated as a part of simulators in several countries.

First, the small break LOCA prediction capability of the large system codes RELAP5 and TRAC, and the fast running SMABRE were compared in the Inter-Nordic SÄK-3 project in 1984–1986 [65]. The SMABRE validation continued against measured data from the LOBI [66] and LOFT [67] facilities. The validation cases are listed in Table 3, and the nodalizations of six integral facilities for SMABRE are shown in Figure 1.

Table 3. Integral tests in the SMABRE validation.

NPP or facility	Reference plant	Volume Scaling	Experiments carried out
LOFT	Westinghouse	1:50	2.5 % cold leg SBLOCA,
LOBI / Mod1	KWU	1:712	0.4 % cold leg SBLOCA
LOBI / Mod2	KWU	1:712	1.0 % cold leg SBLOCA
PIPER-ONE	GE BWR	1:2200	2.6 % recirculation line break, ISP-21
DOEL	Real plant, Westinghouse	1:1	Real plant SGTR accident, ISP-20
SPES	Westinghouse	1:427	Loss of feed water, with core heatup, ISP-22
ROSA-IV	Westinghouse	1:48	5 % cold leg SBLOCA, ISP-26
PMK	VVER	1:2070	7.4 % cold leg SBLOCA
TMI	Real plant, PWR with OTSG	1:1	Steam line break, OECD/NEA benchmark, point kinetic exercise 1, no measurements

The LOCA experiment in the PIPER-ONE facility for a BWR was the standard problem ISP-21 [68]. The small-scale facility proved to be rather complex for the calculation. The standard problem ISP-20, the only ISP arranged for a real plant, DOEL-2, was a steam generator tube rupture [69]. In this accident one steam generator tube had broken during the plant start up. The organizing of ISPs concerning real plants has problems in delivering sufficiently detailed data on the plant. ISP-20 was organized by delivering the RELAP input deck for the participants. This method has since been used in other benchmarks.

The loss-of-feed water transient was studied in the SPES standard problem ISP-22 by assuming a delayed start-up of the auxiliary feed water pumps leading to the core heat up [70]. The core was cooled again after starting the auxiliary feed water injection. The small break experiment in Hungarian PMK facility has been the only foreign VVER-specific integral facility experiment calculated with SMABRE.

A5 % SBLOCA occurred in the cold leg in the ROSA standard problem ISP-26 [71]. The transient was a rather typical SBLOCA, where the comparison of primary pressure and break mass flow was most important. The transient behavior was rather straightforward. The most important feature of the experiment was the large scale of the facility. In the latest validation case, the first exercise of the PWR MSLB benchmark, the point kinetics of SMABRE were compared with other point kinetic models of system codes [72].

SMABRE got its present form through all these standard problems. Several features were included in the code to describe all the components and automation needed in these mainly experimental facilities, as well as the updating of separate code models. This work continued in the applications as well.

The VVER-related applications of SMABRE have included many transient and accident analyses for VVER-440 and VVER-1000 plants (pressurized thermal shock, SBLOCA, primary to secondary leaks, LOCA). Scoping studies have also been made with SMABRE before RELAP or coupled code analyses. The use of a 5-equation model in APROS, similar to that in SMABRE, has strengthened the experience of applying this type of thermal hydraulic solution system.

The simulator applications of SMABRE are listed in Table 4 [30]. The Loviisa VVER-440 full-scope simulator LOKS was a pioneering effort. The two-phase models of SMABRE combined with the fast running capability necessary in a real time simulation in simulators were elements of the SMABRE success in simulator markets at that time.

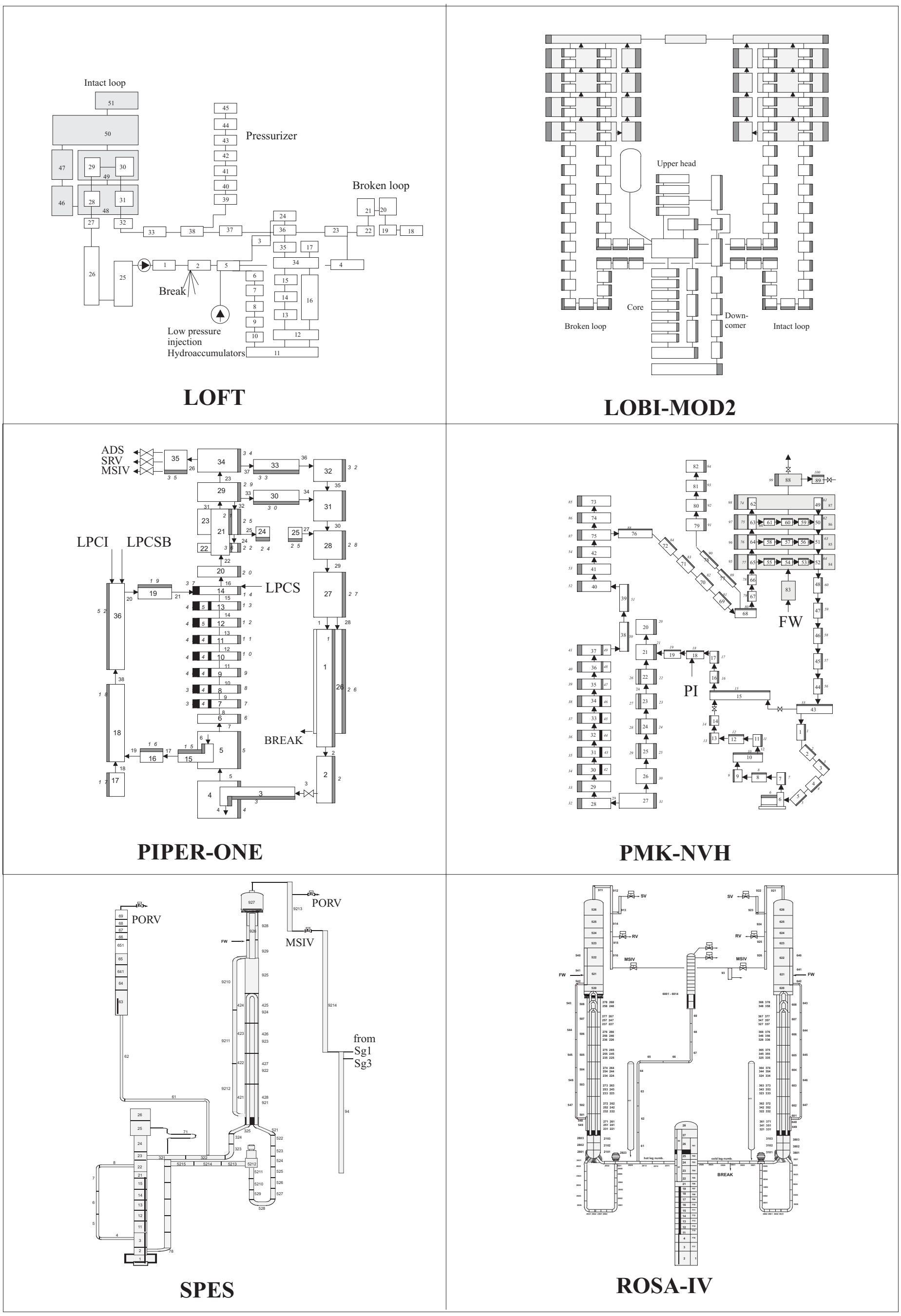

Figure 1. Nodalizations of six test facilities for SMABRE code.

Table 4. SMABRE simulator applications [30].

Interval	Application
1983–1984	Installation as a two-phase model into the Loviisa full scope simulator LOKS.
1987–1988	Installation into the KAERI simulator for the three circulation loop Westinghouse PWR reactor.
1988–1989	Installation into the JAPCO-simulator modeling three plants, one BWR with external circulation, one BWR with jet pumps and one Westinghouse four circulation loop PWR plant.
1988	Installation into the APROS engineering simulator for 5-eq. model.
1990	Installation into two Swedish BWR compact simulators.
1992	Installation into PAKS full scope simulator.

3.2 Validation of HEXTRAN and TRAB-3D codes

The neutron kinetics codes should be validated against experimental data measured at research reactors or real power plants. The shortcomings of the real plant data are insufficient measurements compared with experimental systems. The validation of VTT's neutron dynamics codes, HEXTRAN and TRAB-3D, is summarized in Table 5. The codes include the core thermal hydraulics and TRAB-3D also calculates the thermal hydraulics in the pressure vessel and steam lines for BWRs.

Due to the stepwise progress in the code development at VTT, all the validation work performed for the TRAB and HEXBU codes supports the HEXTRAN and TRAB-3D codes too. The HEXTRAN validation is summarized in [39]. The validation consists of real plant measurements in Loviisa, zero power reactor tests in Czechoslovakia and, the latest, transients in the Moscow V-1000 facility as part of the VALCO project [73]. Here the participants' 3-D neutron kinetics models were validated against a broad spectrum of measurements in the V-1000

Table 5. Validation cases of the HEXTRAN and TRAB-3D core models.

Case	Code	Reactor type
Czech LR-0 test reactor space-time kinetics experiments	HEXTRAN	zero power test reactor with VVER-1000 type fuel assemblies
1st kinetic AER benchmark, control rod ejection	HEXTRAN	VVER-440
2nd kinetic AER benchmark, control rod ejection	HEXTRAN	VVER-440
3rd dynamic AER benchmark, control rod ejection	HEXTRAN	VVER-440, and hot channel
4th dynamic AER benchmark, boron dilution in core	HEXTRAN	VVER-440
Loviisa start-up experiment, reactor trip from 100 % power etc	HEXTRAN	VVER-440
Moscow V-1000 facility transients, control rod movements	HEXTRAN	VVER-1000 zero power reactor with VVER-1000 fuel assemblies
OECD/NEA LWR Core transient benchmarks	TRAB-3D (core)	PWR, BWR
OECD/NEA BWR turbine trip benchmark	TRAB-3D (core)	BWR
Olkiluoto pump trip	TRAB-3D	BWR
Olkiluoto 1 pressurization transient 1985	TRAB-3D	BWR
Olkiluoto 1 instability incident 1987	TRAB-3D	BWR
Olkiluoto 1 load rejection test 1998	TRAB-3D	BWR
OECD/NEA PWR MSLB benchmark in TMI, exercise 2	TRAB-3D	PWR

zero power test facility. With this arrangement the pure neutron kinetics effects may be separated from the feedback effects.

The Atomic Energy Research (AER) group has been a useful forum for the code-to-code validation for HEXTRAN. The first and second kinetics benchmarks, as well as the third and fourth dynamic benchmarks, were calculated with HEXTRAN [74, 75, 76, 77, 78, 79, 80, 81]. The calculation of AER benchmarks was continued with the coupled code validation described in Chapter 3.3.

The validation of TRAB-3D was started with the calculation of the OECD/NEACRP PWR and BWR core benchmarks [40, 82, 83]. The PWR problem was a control rod ejection transient and the BWR problems were a cold water injection and a core pressurization transient. In the second step the circuit models were included for BWR, and a pump trip was recalculated for the Olkiluoto reactor [82]. The international benchmark activity and code-to-code comparisons were continued with the calculation of the OECD/NEA PWR main steam line break benchmark, where exercise 2 was calculated with TRAB-3D. TRAB-3D was coupled to the SMABRE code for the first time in exercise 3 of this benchmark (Paper I). VTT also participated in the calculation of the separate core exercise of the OECD/NEA BWR turbine trip benchmark.

The other line of the validation work has been the comparison of calculated results with plant measurement data. Three such cases, all for the Olkiluoto 1 plant, have been calculated. The first two were the pressurization transient in 1985 and the oscillation incident in 1987 [84]. The most recent validation work covers the calculation of load rejection test from Olkiluoto NPP [85].

3.3 Validation and applications of coupled codes

The best estimate modeling of NPP behavior is part of the modern safety analysis. The codes should be validated, and not just the separate codes but also the coupling of neutron kinetics or dynamics codes with thermal hydraulic codes. Various approaches are possible in this field, but the tests with integral facilities only give limited data due to their typical non-nuclear cores. On the other hand, the test reactors are not designed to meet real thermal hydraulic conditions and the measurements at the facilities do not correspond to the measurements at the real plants.

The organized benchmarks of OECD and AER for coupled codes have been mainly code-to-code validations. Measurements in real plants have been the basis for some international benchmarks, but the specification of the problem and the comparison of results have been closer to code-to-code comparisons where some extreme scenarios have been specified. The already finished Peach Bottom turbine trip benchmark [86] belongs to this category, and the ongoing international benchmark about the VVER-1000 cooling transient in Kozloduy is another example [87]. Using the plant data is challenging because the plant operation is complex, but the available data is not optimal for the validation in all respects.

The first thermal hydraulics/neutron dynamics coupled code at VTT, the 1-D SMATRA code, was validated against real measurements in plants, consisting mainly of Loviisa and Paks startup-tests. The validation cases and used VTT codes are summarized in Table 6. Here the dynamic benchmarks organized by AER and OECD are code-to-code comparisons.

The HEXTRAN-SMABRE coupled code has been extensively applied to the Loviisa VVER-440 safety analyses since the 1990s [40, 88 and Paper V], to the Hungarian VVER-440 type Paks NPP [89], to the preliminary analysis of the VVER-91 concept designed for Finland at the beginning of the 1990s [90], and later to the Kola and China VVER-91 projects in co-operation with IVO International. The HEXTRAN-SMABRE safety applications consist of all the main transient types: loss of feed water (LOFW), control rod ejection (CRE) or withdrawal (CRW), loss-of-offsite power (LOOP), one or two steam line breaks (MSLB) and pump seizure, including several ATWS transients, and boron dilution cases for VVER-440 and VVER-1000 – some of them carried out in start-up or cold conditions.

Table 6. Validation cases for coupled codes at VTT.

Case	Code system	Reactor type	Ref.
Loviisa NPP - 2 of 6 RCP trip - 1 of 2 turbine trip - overcooling transient - reactor trip	SMATRA	VVER-440	[49, 50]
Paks NPP - 1 of 6 MCP trip - reactor trip	SMATRA	VVER-440	[91]
Main steam header break, AER BM5	HEXTRAN-SMABRE	VVER-440	[8, 92]
Main steam line break, AER BM6	HEXTRAN-SMABRE	VVER-440	[9, 93]
OECD PWR MSLB benchmark in TMI, exercise 3	TRAB-3D-SMABRE	PWR	Paper I
Load drop to in-house power level in Loviisa	HEXTRAN-SMABRE	VVER-440	Paper II
Feed water pump trip in Balakovo	HEXTRAN-SMABRE	VVER-1000	Paper III
Control rod drop in Bohunice	HEXTRAN-SMABRE	VVER-440	Paper IV
1 of 3 RCP trip in Kozloduy	HEXTRAN-SMABRE	VVER-1000	Paper IV

3.3.1 MSLB in TMI with TRAB-3D-SMABRE, Paper I

The pressurized water reactor Main Steam Line Break (MSLB) benchmark transient using the Three Mile Island Unit 1 (TMI-1) nuclear power plant as the reference plant provides an international code-to-code [94] benchmark for the coupling of 3-D neutronics codes to thermal hydraulics system codes. The TMI-1 plant has two coolant loops, and a steam line break in one of them leads to an asymmetric cooling of the reactor core. This, together with the assumption of one stuck control rod, makes this numerical benchmark interesting for the 3-D analysis of the core. Of course, no such plant transient data exist and this benchmark is, therefore, a code-to-code comparison.

The benchmark calculation was divided into three exercises, so that separate testing of the core and plant models was possible. The first exercise included the full plant model and point-kinetic neutronics. The second exercise consisted of the separate 3-D neutronics model of the core using the specified hydraulic boundary conditions. In the third exercise the plant and the core models of the first two exercises were coupled together.

The definition of the plant geometry as a system code input and the nuclear data given in the specifications were helpful in the estimation. On the other hand, the duration of this benchmark with three exercises was several years. Altogether, 17 calculations were supplied in exercise 3. For example, SMABRE, RELAP5, CATHARE, ATHLET and TRAC were used as the system codes with several kinds of couplings to neutron kinetics codes. Comparison of the results from exercise 1 and 3 demonstrates that the 3-D neutron kinetics may remove some of the conservatism inherent in the point kinetics [95].

The first and second exercises of the benchmark were calculated with the SMABRE [72] and TRAB-3D codes respectively. For the third exercise these two codes were coupled together for the first time, though the same coupling method had been used in HEXTRAN-SMABRE with the hexagonal fuel geometry since the early 1990s.

In the third exercise the possible occurrence and timing of the recriticality after the reactor scram, caused by the continuing reduction in the core inlet temperature, was highly dependent on the modeling of the primary and secondary hydraulic circuits.

The steam generator of the TMI plant is a once-through model. Unlike the other vertical SGs, the heat transfer tubes are straight vertical tubes, secondary water is superheated, and a weak circulation exists in the SG due to steam flow in an aspirator junction used for the preheating of the feed water. The behavior of this kind of SG in accidents is not straightforward.

Several variations concerning the nodalization and steam generator details, as well as the mixing in the primary side, were calculated with the SMABRE model as a part of the first exercise. Some of these cases were recalculated using the coupled code and the specifications for the third exercise. The effects of two secondary circuit modeling variations dealing with the separation of phases in the steam lines and the flow reversal in the aspirator junction are presented in Paper I, and in [96] and [97].

The VTT calculations showed that with the two variations of the secondary side modeling presented in Paper I it was possible to cover nearly the whole spectrum of results calculated by the participants in the MSLB benchmark. The exercise clearly shows that the uncertainties in the thermal hydraulics were much larger than in neutron kinetics when the nuclear data was given.

Similar conclusions can be outlined from the results of all the participants in this benchmark. In the first exercise the results of several system codes clearly deviated. The results of the second exercise, with the separate 3-D neutronics model of the core using the specified hydraulic boundary conditions, were in very good agreement. Further, the third exercise, using the plant and the core models of the first two exercises coupled together, showed similar deviations between the participants' results to those of the first exercise.

3.3.2 MSLB with HEXTRAN-SMABRE for VVER-440

VTT participated in the code-to-code validation in the AER dynamic benchmarks with the coupled code HEXTRAN-SMABRE [8, 9]. The 5th AER dynamic benchmark, a symmetrical over-cooling transient at hot standby of VVER-440, was the first AER benchmark for coupled 3-D neutron kinetic codes and thermal hydraulic codes. The second, the 6th AER benchmark, was an asymmetrical steam line break at the end of the first fuel cycle under the full power conditions of VVER-440. The participants in the AER benchmarks, research centers in Finland, Germany, Hungary, Czechoslovakia and Russia, were supposed to have VVER-models of their own and, unlike the OECD/NEA standard problems, only the main features were specified. It was assumed that each participant applied their own nuclear data, own core and steam generator models. Several conservative assumptions were added compared with the best estimate safety analyses.

The plant model for SMABRE is mainly based on two input models, the Loviisa model and the standard VVER-440/213 plant model. The main differences between these two plants, to be taken into account in modeling, are listed in Table 7. Small differences inside the pressure vessel are taken into account just by tuning the loss coefficients. In the SMABRE model the reactor pressure vessel is divided into six parallel sectors. Orientation of the loops is described in Chapter 5.2. Due to the symmetry in the 5th benchmark, the 1/6 core model was used in the HEXTRAN model for the core. In the 6th benchmark the whole core had to be modeled due to the asymmetry in the core [98].

The steam generator model has an important role in steam line breaks. The horizontal steam generators were modeled for SMABRE with several layers as described in Chapter 5.1.

In the 5th benchmark the overcooling in the transient is created as a consequence of the main steam header break at the hot standby state. When the break opens, the pressures in the steam generators drop rapidly. The liquid temperature decreases symmetrically in the core inlet leading to the return to power. No isolations of the steam generators were assumed. In the HEXTRAN-SMABRE calculation the maximum core power was about 38 % of the nominal power at four minutes after the break opening. The core power is decreased by boric acid in the high pressure safety injection water and the reactor is kept subcritical from this moment forward.

The calculated nuclear powers of the participants in this AER benchmark are depicted in Figure 2. In addition to the correct thermal hydraulic modeling of the phenomena, accurate nuclear data is important in the calculation of this kind of cooling transient, where changes in the temperatures and densities are large and recriticality is achieved. Large differences were found between the participants' results, even though the initial reactivity level was defined in the benchmark specification [92]. In the VTT calculations nuclear data was based on the ENDF/B-IV library and was evaluated with the CASMO-HEX code. The importance of the nuclear data was illustrated by repeating the benchmark calculation in VTT with three different data sets [8]. The recently added wide range cross section model [99] in HEXTRAN was not available.

Table 7. The main differences between the standard VVER440/213 and the Loviisa plant (from the thermal hydraulic modeling point of view).

1.	In Loviisa the bottom of the pressurizer is below the hot leg nozzles. The two surge lines are connecting the pressurizer to the hot legs of two loops before the loop seals. In the standard plants one surge line is connected to the loop seal of one hot leg.
2.	In Loviisa the main circulation pumps are Finnish Ahlstrom-made pumps where the suction takes place from the side and the discharge is downwards. In the standard plants the discharge is on the opposite side of the suction point.
3.	The geometry of the cold leg and hot leg, including the lengths and heights of the loop seals, is different in Loviisa mainly due to the two reasons above. The differences are shown in Figures 3 and 6 with approximate shapes of the circulation loops.
4.	36 dummy steal assemblies are used in the peripheral area of the Loviisa core, decreasing the active core flow area.
5.	Loviisa has its own protection signals, trips and control logic, slightly modified over the years.
6.	The nominal power level is now 109 % in Loviisa, corresponding to 1500 MW thermal power.
7.	Numerous differences exist in operating and safety systems, such as pipeline connections and injection points in makeup and safety injection systems. In the secondary side of Loviisa feed water is injected above the steam generator tube bundle, whereas the injection point has originally been inside the tube bundle.

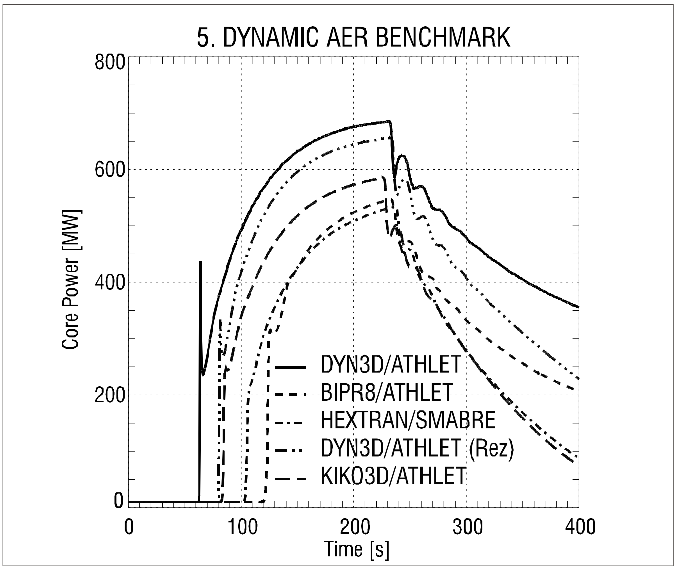

Figure 2. Calculated total core power by five participants in the AER 5th benchmark [92].

Large differences in the fuel heat transfer models were generated between the maximum fuel temperatures among the calculations of the participants already in the 5th benchmark.

The AER 6th benchmark concerns the double ended break in the main steam line in a VVER 440 plant. The core is at the end of its first cycle under full power conditions. In the sequence, the main steam line break at the nominal power level is followed by the reactor trip. The closure of the main steam isolation valves isolates the broken loop steam generator from the others and the liquid temperature continues decreasing only in one core inlet sector, which may lead to recriticality and neutron power increase. The resulting power level is maximized by the assumption that the main circulating pumps are not stopped and, hypothetically, two control rods are stuck in the uppermost position. Flow mixing in the pressure vessel before the core was assumed to be 30 %. The isothermal recriticality temperature was tuned to be high, 210 °C, in the calculation. Therefore, the calculated cases here were almost hypothetical. Only a mild power increase was seen in the VTT calculations.

Five coupled code calculations were compared in the 6th benchmark [93]. The results deviated, especially at the time of the possible recriticality. The main reasons originated from the thermal hydraulics on the secondary side and the differences in the steam generator models. Here, either a one-dimensional secondary side model of the steam generator or two channels models with internal circulation were used. Further, the modeling of the five vertical tubes from the steam generator outlet to the steam lines was found to be important for phase separation. The comparison of the results with MSLBs in the other plants is done in Chapter 5.5.

4. Use of real plant data for validation

In this chapter, the essential part of this thesis, the coupled code validation, is continued against the real plant data. The work has been performed in international projects. The author had main role in the plant models and calculations with SMABRE. She also participated in the problem specification and evaluation of the results.

The data for the code validation consists of the experimental data from the test facilities, numerical benchmarks and data measured in real nuclear power plants. Here a valuable addition has been the European Union projects, where several transients at real VVER plants have been documented and made available to all participants for validation purposes. Five transients, three measured at VVER-1000 and two at VVER-440, were documented in the PHARE project SRR1/95 [23, 24]. One transient was chosen for the code validation for both VVER types. Here the comparison of a calculation not just with the measurements but also with the other calculations by the other participants gives additional value for the evaluation of results. The results are reported in Papers II and III.

In the EU/VALCO project [25] the data collection was extended to new types of transients and the validation of coupled codes was continued (Paper IV).

4.1 Turbine trip in Loviisa, Paper II

Seven countries participated in the validation of coupled codes in the SRR1/95 project. The thermal hydraulic system codes were ATHLET and SMABRE. The 3-D neutron kinetics codes were HEXTRAN, the German DYN3D [100], the Hungarian KIKO3D [101] and the Russian BIPR8 [102] – altogether, four coupled codes.

Five real VVER plant transients were documented in the first phase of SRR1 and two were selected for the coupled code validation. The calculated VVER-440 transient was ‘Turbo-generator load drop experiment at Loviisa-1’. The experiment was performed just after the plant modernization and, as part of start-up tests, more measured data, especially from the first minutes, was available for validation than during the normal operation of a NPP. The utility supplied the test report and, for the core, burnup and neutron power distributions, evaluated with HEXBU-3D. The HEXTRAN-SMABRE model of the Loviisa plant extended to the turbines, whereas in the ATHLET model the secondary pressure was used as a boundary condition. All the calculations were performed without the models of power controllers. The boundary condition on the positions of control rods was defined by the participants based on some details recorded at the plant. The calculated results were reported and sent to Finland for the comparison.

4.1.1 Comparison of calculations

A very good accuracy of the results was generally achieved in all the calculations. The core and primary circuit parameters were within the measurement accuracy. The primary pressure behavior was sensitive to the operation of the pressurizer spray and heaters. It was shown that the coupled neutron kinetic / thermal hydraulic codes under consideration were capable of simulating this type of VVER plant transient. The systematic and detailed data collection, as well as the good reporting quality performed by the utility, played an important role in this validation effort.

In all the calculations with 1-D thermal hydraulics, the unmixed water (see Chapter 5.2) in the upper head of the VVER-440 pressure vessel resulted in an almost unchanged water temperature in spite of a reduction in pressure. For the same reason, slightly higher primary pressure and pressurizer levels were obtained than in the measurements. The model with mixing junctions – i.e., flow connections from the upper head to the hot legs – was demonstrated. This artificial treatment of the mixing problem could be avoided if, according to the conclusions, some parts of the cooling circuits, e.g. the lower and upper plenum, were modeled three-dimensionally. In the VVER type plants the modeling of the horizontal steam generators would also largely benefit from 3-D modeling.

The simulation of the time delays in the measuring devices was developed and successfully demonstrated for the core outlet temperature measurements and signals of the self-power neutron detectors (SPND). The neutron detector readings of SPND were compared with the calculated thermal flux, taking the detector time constants into account. The agreement was good in all the calculations, but perhaps the most significant result was the demonstration of the suitability of these measurements for validation in spite of the large time constants.

A large time constant of 30s was determined by comparing the measurements and the calculations of the core outlet temperatures. This approach was justified for the coolant temperatures because of the good agreement between the calculations by different codes. The time constant was later confirmed by the utility. A time constant of 10 s was used for the loop temperatures, in accordance with earlier studies [50].

Some deviations between the results and measurements can be explained by uncertainties in the measurements. Of course, it is important to take account of the systematic errors of the measurements whenever possible. When the real plant is involved, the absence of the exact mass flow measurements is a handicap, especially for the core simulation. Typically, the mass flows are converted from pressure difference measurements. Further, due to the small number of measurements and the inadequacy of the plant modeling, the results are somewhat dependent on the way each participant decodes the real plant data. This is clearly seen in defining the control rod positions by participants during the transient. Also, there are several other factors affecting the fission power level during the power reduction in the transient; the control rod reactivity worth (efficiency) and the uncertainties in the nuclear data affect both control rod efficiency and Doppler feedback, as well as the uncertainties in the fuel temperature calculation. All these effects are difficult to separate.

According to the recommendations of the project, in order to separately studying the control rod efficiencies the neutronic codes, including the applied nuclear databases, should be additionally validated against measurements at real plants or research reactors under zero-power conditions. This was realized in the next VALCO project for a VVER-1000 core [72]. In order to decrease the large deviations in fuel temperatures, a more detailed gas gap modeling, and, further, a dynamic modeling of the gap width, were required in the conclusions of the SRR1 project.

4.1.2 HEXTRAN-SMABRE calculation of Loviisa case

Few changes were made to the Loviisa plant model of HEXTRAN-SMABRE for this case. The modifications mainly considered the secondary side modeling in order to produce the proper boundary condition for the 3-D dynamic core calculation.

In SMABRE, the primary side included six separate circulation loops, one of which is depicted in Figure 3. Excluding the upper head, the pressure vessel was divided into six vertical sectors. The flows between these sectors mainly define mixing in the pressure vessel before the core. The mixing junctions described in the previous chapter were used for the upper head mixing. The size of the mixing junctions was studied in the Loviisa case and, as a result, 3 % of the primary flow was needed to mix the upper head water.

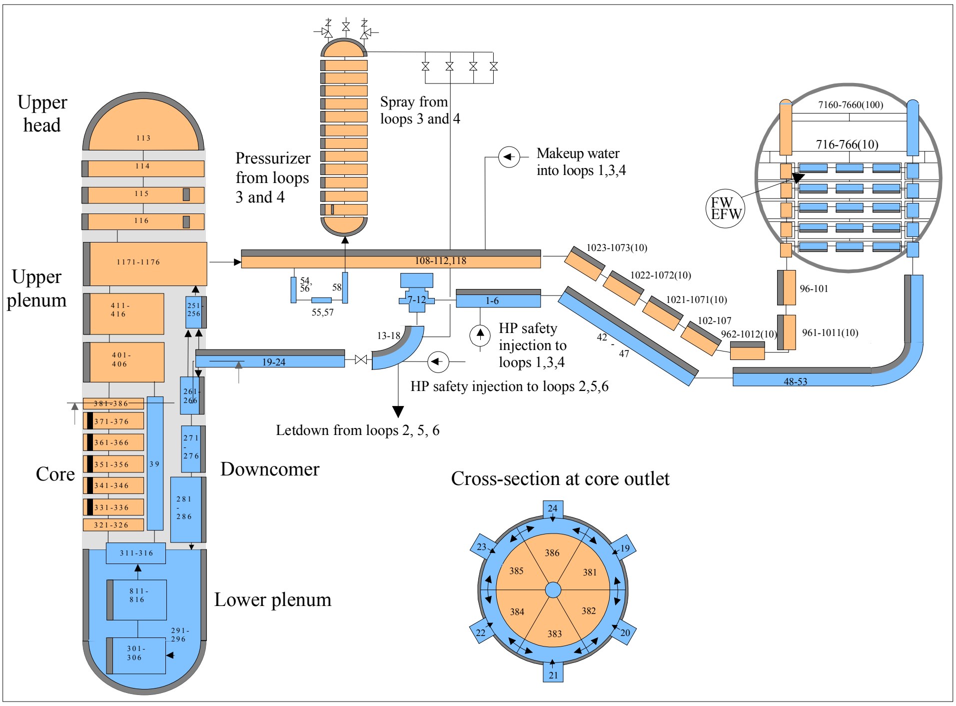

Figure 3. Nodalization of the Loviisa primary side for the SMABRE code.

In HEXTRAN, a 60-degree symmetry sector of the reactor core was applied in the transient calculation and each fuel assembly in the sector was described by a separate hydraulic channel. Above the hydraulic channels, the mixing of the core by-pass flow with the heated mass flow through the fuel assemblies before the temperature measurements in the plant was not taken into account in the calculated values. Therefore, the temperature distribution at the core outlet, calculated with HEXTRAN, resulted in slightly higher values than the measured temperatures

The steam generator model was originally created for RELAP5 [103], and about the same model is used in SMABRE and APROS. The main principle in the nodalization was to enable primary and secondary water circulation inside the steam generator. Further, the riser part was clearly more voided than the downcomer part in the SMABRE model. The feed water lines were described up to the feed water tank, as seen in Figure 9 on page 71. The two feed water tanks in the plant, with two parallel lines, high pressure heaters and valves, were described with one line in the model.

The calculated initial fission power distribution was compared with the measurements. Some axial tilt was seen, the calculated power at the bottom part of the core being slightly larger than the measured values. Further, the calculated and measured neutron power occasionally deviated during the transient. One possible reason for the differences was the location of the out-of-core neutron detectors axially in the middle of the core.

4.2 Feed water pump trip in Balakovo, Paper III

In the EU Phare project SRR1/95 the coupled codes were validated against a transient at a real VVER-1000 plant. The transient was ‘turn-off of one from two working SG feed water pumps at Balakovo-4’ in Russia. The thermal hydraulic system codes were ATHLET and SMABRE. The 3-D neutron kinetics nodal codes were HEXTRAN, DYN3D and BIPR8 – altogether, three coupled codes at five institutes. Three codes were used for the nuclear data generation.

4.2.1 Comparison of calculations

Thermal hydraulics was solved one-dimensionally in all the codes and the flow channels in the core were solved as separate channels, although the VVER-1000 core without shrouds around fuel assemblies necessitated cross flow modeling between channels, at least in the best estimate calculations. The initial event being on the secondary side, the secondary pressure as a time-dependent boundary condition was used in the ATHLET model. In SMABRE, the secondary pressure was the result of the heat transfer from the primary side, the feed water pumps operations and the controller modeled for the turbine valve.

In the test, one of the two working steam generator feed water pumps was switched off at the nominal power. The power control system responded to the pump trip by control rod actions. In the secondary circuit the feed water flow decreased dramatically at the feed water pump trip. Due to the power drop in the core, the need for feed water dropped. The steam generator water levels only decreased slightly, the heat transfer tubes remaining below the water level during the whole transient.

The result of the validation for all the coupled codes involved was successful. In the main, a good agreement was achieved between the calculated and the measured safety-relevant parameters. The main outputs from the neutron kinetic codes, the fission power and the power distributions were quite accurate, which express the main thermal hydraulic parameters as sufficiently accurate for the calculation of feedback effects. On the other hand, the behavior of the control rods was a boundary condition in all calculations. Further, as part of the project, the importance of the dynamic heat transfer coefficient in the gas gap was demonstrated.

The calculated primary pressure and the pressurizer water level compared quite well with the measured data, but the sudden reduction in both of these values was not large enough in all calculations compared with the measurements. Typically, both the letdown and makeup are operating in real plants, even under the nominal state. Further, the constant operation of a few pressurizer heaters may be needed to compensate the heat losses from the primary side. A small spray may also be operating continuously. The operations of all these systems were not realistically simulated in the calculations, partly due to the lack of information. The effects of the pressurizer heaters were illustrated by the calculation of one variation.

As a result of these validations, and against the Balakovo data, a strong recommendation could be given for updating the fuel models of the coupled codes due to the large deviations in fuel temperatures between calculations. This contains several possible details: a dynamic gas gap width; burnup and temperature dependence of all fuel rod materials; radial burnup and density distribution in fuel rods; and internal rod pressure. Many of these points are already modeled in the special fuel behavior transient codes.

As a recommendation for future coupled code validation, the uncertainty analyses were proposed in order to study the quantitative influence of several sources of uncertainty – e.g. in the model parameters, reactor operating conditions and systems.

4.2.2 HEXTRAN-SMABRE calculation of the Balakovo case

The HEXTRAN-SMABRE model of VVER-1000 is the result of several applications. The first basis of the model was the VVER-91 concept for Finland at the beginning of the 1990s. This Russian-type VVER-1000 is currently under construction in China. The model was improved step by step in applications dealing with other VVER-1000 plants in Russia. Thereafter, the Balakovo model was created with a few changes in the SRR1 project.

The primary circuit nodalization, about the same as for Kozloduy in Figure 7 on page 58, shows that all four loops were separately modeled, including the steam generators, reactor coolant pumps, hot legs and cold legs. The pressurizer is connected to the hot leg of loop 4.

In the Balakovo model the reactor vessel below the hot leg elevation was vertically divided into four parallel sectors according to the number of circulation loops. Further, all volumes in the downcomer, lower plenum, core and upper plenum were divided into four volumes with horizontal junctions between them. Special cross flow junctions of SMABRE were defined in the downcomer and lower plenum horizontal junctions. A single node represented the coolant flow bypassing the core inside the control rod guide tubes and reactor baffle. Another node bypassed the upper plenum from the core exit to the upper head. In HEXTRAN, a 60-degree symmetry sector of the reactor core was applied in the transient calculation and each fuel assembly in the sector was described with a separate hydraulic channel. All the HEXTRAN calculations were performed with 20 axial nodes for the active height of the core.

The general secondary side nodalization is shown in Figure 4. The steam generators were modeled according to the same principles as used for the steam generator of VVER-440s [103]. The liquid volume in the secondary side below the nominal water level, as well as the steam dome, was vertically divided into five nodes in order to better describe the phase separation.

The water volume surrounding the tube bundles was modeled as a separate downcomer. The model enabled primary and secondary circulation and, on the secondary side, a more voided riser volume compared with the downcomer volume.

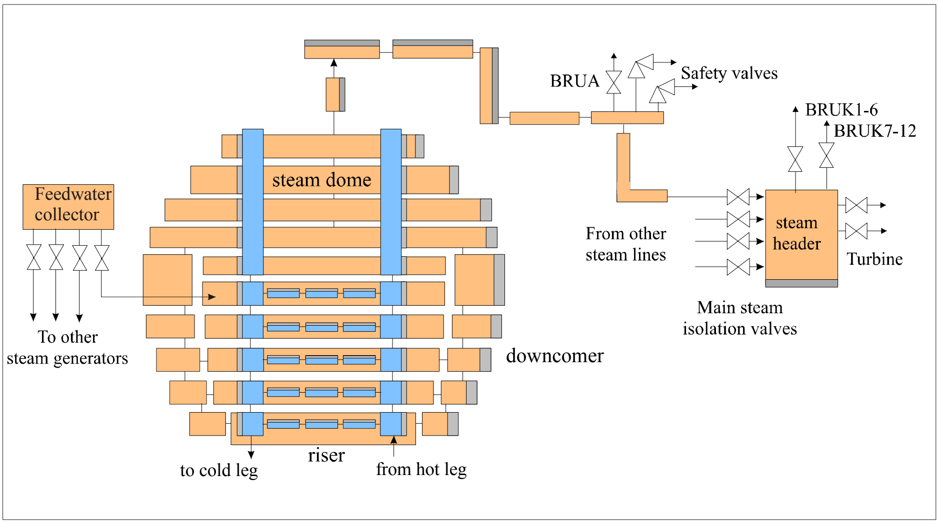

Figure 4. Balakovo VVER-1000 secondary nodalization for SMABRE code.

For the Balakovo transient, care was taken to model the secondary controls, the feed water and the turbine control. The feed water flow and the feed water-steam flow difference, together with the steam header pressure behavior, were considered in the turbine valve control.

At the end of the transient there were differences between the calculated and the measured data, indicating that the neutron power level stayed too high in the calculations. The calculated power level was the same as that obtained in the neutron flux measurements, but higher than the measured thermal power based on the thermal hydraulic measurements. A possible explanation for the higher values of calculated power when the reactor approaches the new steady state after the transient was the neglecting of xenon dynamics.

In the HEXTRAN-SMABRE calculations an assumed time constant of 16 s for temperature measurement gave the best result in the comparison of the calculated loop temperatures with the measured values.

4.3 CR drop in Bohunize and RCP trip in Kozloduy, Paper IV

The successful previous PHARE project SRR1/95 was an initiation into the EU project VALCO, where the collection of data on real plants was continued and the validation of coupled codes was extended to new types of transients. Besides, and according to the recommendations in the SRR1/95 project, the neutron kinetic codes were validated separately against measured data [72], and the uncertainty and sensitivity analyses were performed for the transients already analyzed in the SRR1/95 project [Paper VII].

Data was collected on five real plant transients at the beginning of the VALCO project, three concerning VVER-440 plants and two VVER-1000 plants. Two of them, ‘Drop of control rod at nominal power at Bohunice-3’ in Slovakia for VVER-440 reactors and ‘Coast-down of 1 from 3 working MCPs at Kozloduy-6’ in Bulgaria for VVER-1000 reactors were then chosen for code validation. The former is an unexpected event focusing on asymmetric core power and pressure vessel mixing phenomena, whereas the latter is part of the plant startup tests and emphasizes loop thermal hydraulics. The initial events of these transients are primary side related, whereas they were secondary side related in SRR1/95.

In VALCO, eight institutes participated in the code validation with five different coupled codes and ten calculations. Six teams applied ATHLET as the thermal hydraulic code; RELAP5 and SMABRE were also used. Five teams used DYN3D as the neutronics code, in four teams coupled to ATHLET. The other neutron kinetics codes were HEXTRAN, KIKO3D and BIPR8.

The validation was successful for both cases and the results were reasonably accurate, but not without difficulties. Due to partly parallel data collection and validation efforts, the data report could be complemented with necessary details, but, on the other hand, the usable time for calculations shortened respectively. Typically, only one calculation round was possible. The task emphasized careful plant data interpretation and balanced plant modeling, where transient-specific asymmetric phenomena played a key role. Due to the asymmetry, the whole core model was needed in the calculations in the Bohunice case and the separate modeling of all the four circulation loops in the Kozloduy case.

In the following, both cases are discussed in more detail, and also through VTT's contribution to the validation of HEXTRAN-SMABRE.

4.3.1 Comparison of Bohunice calculations

In the Bohunice case the unexpected control rod drop was followed by the actions of the control rods. The burnup and xenon distribution were specified in the data report, but their simulation according to the detailed operation history was also possible. Only this first phase, lasting a reasonably long time, was calculated and compared with the measurements. Due to the uncertainty in indicating the actual positions of the control rods in VVER440s, two slow actions were specified to simulate the measurements. The final power level was also predicted at 85 %, as well as the applied time constants in the measurements. The large deviations between the calculations of power level after the rod drop, the axial power profile and the calculated control rod would seem to indicate that fuel modeling and treatment of VVER-440 control rods needs further consideration.

The calculated fission powers were compared with the measured neutron power. The calculated powers before the control rod actions dropped more than the measured but showed similar behavior. In the highly asymmetric Bohunice case the measurements from the out-of-core detectors and the readings of thermocouples at the core outlet were successfully exploited in the validation.

The signals from the rhodium SPND detectors were used to confirm the 3-D power distribution, although no explanation of the suitable time constants needed for comparing the calculated values with the measured ones was found, see also Chapter 5.3. The time constants of integration for the delayed part of the signals were clearly different to those in Loviisa. Figure 5 gives an impression of the total number of the measured and calculated SPND signals compared in the project. The original shapes of the calculated SPND signals without the simulation of time delays closely follow the simulation of the out-of-core detector signals (Figure 2 of Paper IV).

In the Bohunice case the elementary modeling of coolant mixing in the reactor pressure vessel improved the correspondence to the measurements. The result, according to the hot leg temperatures, indicated a rather weak mixing in the upper plenum.

4.3.2 HEXTRAN-SMABRE calculation of Bohunice case

The SMABRE model for Bohunice Unit 3 is a combination of the standard VVER440/213 model and the Loviisa model. It is described in context with the AER benchmarks in Chapter 3. The primary side nodalization is shown in Figure 6. Similar to the AER 6th benchmark, 30 % mixing was assumed between the sectors before the core inlet, and, as in the AER model, the SMABRE sectors were rotated 30 degrees compared to the Loviisa model – see Chapter 5.2. This modeling seems to fit well with the measured hot leg temperatures. No mixing was modeled in the upper plenum above the core. The whole core model was used for the Bohunice core in the HEXTRAN model. All the 349 fluid channels were individually modeled.

The secondary side was modeled with the steam generator taken from the Loviisa model for SMABRE, but the rest of the secondary side was rather simple. In the primary side the model was adjusted for Bohunice with friction coefficients in order to fix mass flows, pressure drops, bypass flow rates, etc.

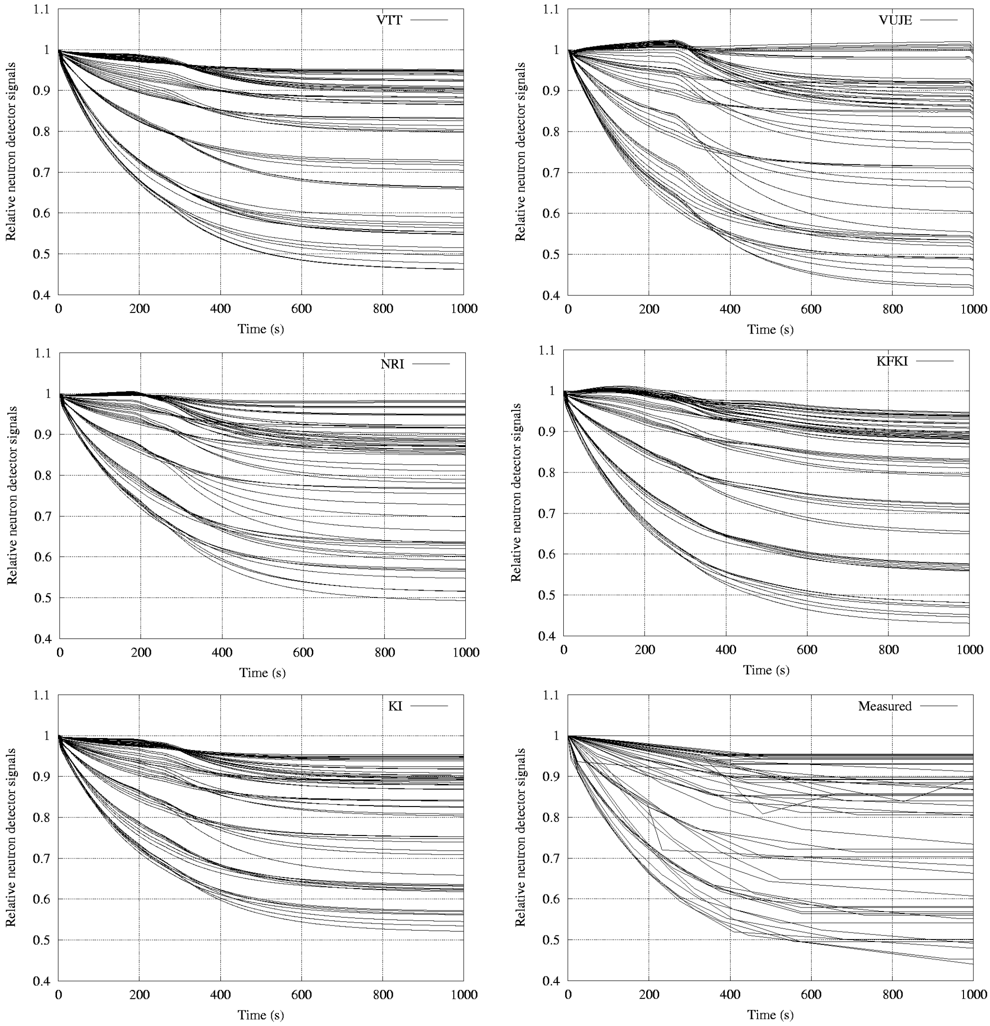

Figure 5. Measured and simulated SPND signals by five institutes from seven elevations of eight fuel assemblies in the core in the Bohunice rod drop case [104]. All signals have been scaled to the value 1 at the beginning of the transient.

According to the data reports prepared by the utility, the core bypass flow – totally 9.12 % of the flow to the pressure vessel – consists of several components. The main part is the flow getting out through the holes in the assembly lower part and getting back through the holes in the assembly upper part before the core outlet temperatures are measured. For HEXTRAN, the bypass flow through these holes was not modeled.

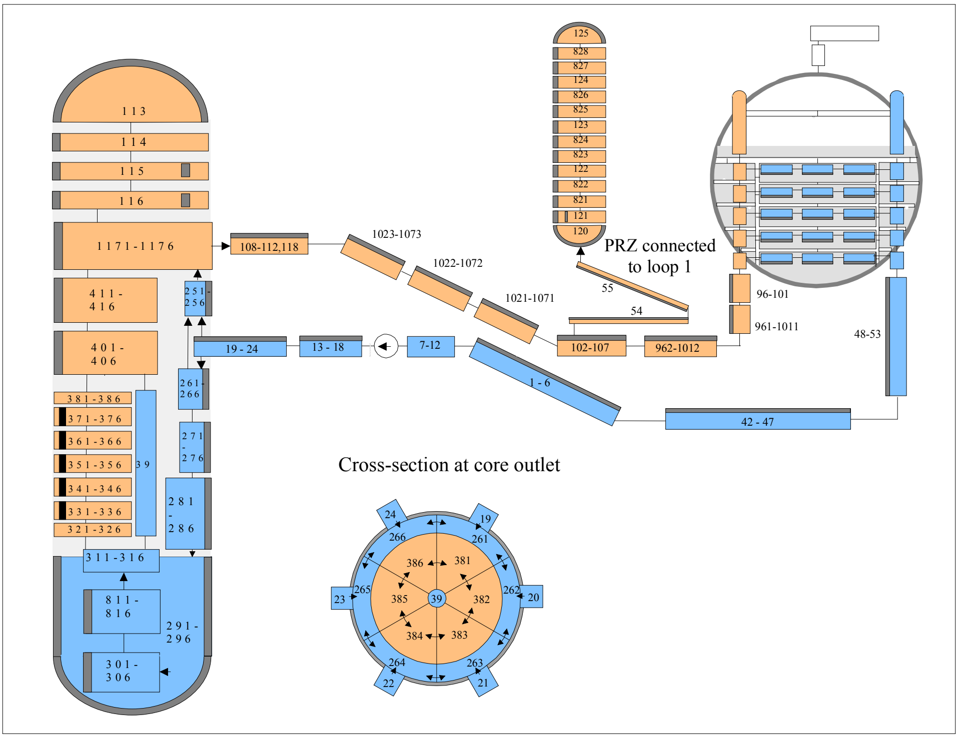

Figure 6. Bohunice primary side nodalization for SMABRE code.

4.3.3 Comparison of Kozloduy calculations

The Kozloduy test was measured in 1992, thus tracing some details in the measurements was not successful, although two transients were documented in the data collection. The main coolant pump tripped at the nominal power, and the second pump tripped at 65 % power level one and a half hours later. The later trip was used in the validation. The steam header pressure and the position of the regulation control rod group were used as boundary conditions in several calculations, but the option of the power controller was also used.

The features that make the Kozloduy transient interesting, such as the lowered power at the beginning of the test and the flow reversals in the loops, proved to be difficult both for data collection and for modeling. As an example, it was hard to find detailed data on the pump characteristics and control logics. The difficulties with the pump curves appeared again in the context of the VVER-1000 cooling benchmark organized by OECD [105]. Anyway, the general behavior of the Kozloduy second pump trip was calculated satisfactorily with all the codes and the large number of measurements could be used successfully in the validation.

In the comparison of the core outlet temperatures, a slight linear dependency was found between the assembly power and the difference between the measured and the calculated temperatures. The dependency could possibly be explained by the temperature effect of the bundle central tube flow. The difference was proportional to the total mass flow used in the calculations, and quite different mass flows were used by the participants. It was difficult to estimate the correct mass flow partly due to difficulties in defining the pump curves and the inadequate pressure differences over the main coolant pumps in the calculations.

However, the real plant data has its problems in the validation. The correct interpretation of plant measurements is important. The unmeasured mass flows, the non-specified core bypasses, and the deviations between the real and the reported set points of operating systems are typical difficulties in utilizing inevitably deficient real plant data. This supports the code-to-code validation to a certain extent.

In the Bohunice and Kozloduy cases the calculated initial fuel temperatures and the temperature changes during the transient varied remarkably. This supports the conclusion of the previous SRR1/95 project that more accurate fuel models are needed in the codes. The need for development in reactor dynamics is also reported in the CRISSUE project [41] performed in co-operation with the VALCO project. The remaining transients, measured in VVER-1000 and VVER-440 plants and documented in the SRR1/95 and VALCO projects but not yet calculated with the coupled codes, could be used for further validation work.

4.3.4 HEXTRAN-SMABRE calculation of Kozloduy case

The input model for Kozloduy VVER-1000 is based on the input for the Balakovo VVER-1000 plant in the SRR1/95 project. The nodalization scheme for the primary loop is shown in Figure 7. The secondary side, shown in Figure 4, is the same as that used for the Balakovo case.

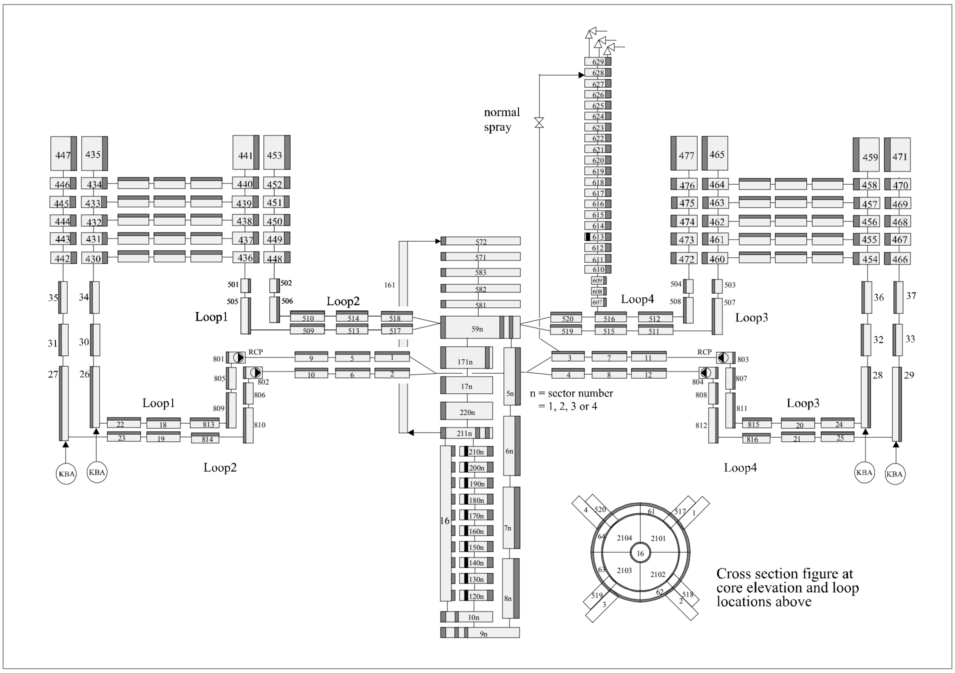

Figure 7. Kozloduy VVER-1000 primary side nodalization for SMABRE code.

The nominal state calculated with the Balakovo model indicated several differences in nominal parameters from the Kozloduy measurements. Therefore, in the Kozloduy nominal state calculation the mass flow rates of the primary circuit loops were adjusted with the pressure drops over the core and the reactor vessel. The pressurizer level control was modeled according to the average temperature of the operating loops. The controls of the secondary side were simplified.

In the first calculations the mass flow rate in the switched-off loop seemed to differ significantly from the value reported in the data collection report. The homologous pump curves were updated using several references – which, by the way, also proved to be inconsistent. The nodalization of the loops was made more detailed. The pump characteristics are typically given with homologous curves in eight sections in system codes. The sections correspond to pump and turbine operation with forward and reversed rotation. The curves given by the plant represented the typical pumps for VVER-1000, but the given mass flows and pressure differences in Kozloduy were somewhat different. Further, the pressure differences over the main coolant pumps seemed to differ from the measurements in the HEXTRAN-SMABRE calculation. Typically, this kind of deviation exists when the modeled primary side loop geometry differs from the real geometry or the distribution of pressure losses between the pressure vessel and the circulation loops are inconsistent.

The initial core of Kozloduy 6 consisted of fuel assemblies having several enrichments. The fuel assembly burnup program CASMO-4 was used to create the nuclear cross sections. The burnup distribution of the first cycle of the Kozloduy 6 reactor was simulated with HEXBU-3D.

According to the detailed operation history, the reactor power was close to full power for 40 hours before the coast-down of the first pump. This resulted in a equilibrium xenon poisoning in the core. The power reduction after the first pump trip started a xenon transient of some significance. This was not considered in VTT's calculations.

Furthermore, a time constant of 10 seconds was assumed for the temperature measurements in order to compare the measured hot and cold leg temperatures with the calculated values. These simulated hot and cold leg temperatures were quite compatible with the measured data. The calculated pressure drops over the main coolant pumps and over the pressure vessel showed correct time behavior, even though the measured values for the pumps were not reached.

5. Application of codes

This chapter presents the several issues that emerged in applying the coupled codes in Papers I–V of the thesis and earlier applications. The several methods used, partly developed by the author, are described. In the example of safety analyses, the author prepared the SMABRE part of the Loviisa case. At the end, she draws some conclusions from the steam line breaks dealt with in this thesis.

The final use of codes may vary from one application to another. For the licensing of a reactor, the margins of safety are the most interesting issues. In the planning of the economical use of a reactor, or in the simulator use, the target may be partly different. The best estimate, BE, calculation is the basis for modern safety analyses, and the desired conservatism is typically brought in separate hot channel calculations or with sensitivity and uncertainty analyses, as discussed in Chapter 7.

5.1 Nodalization questions in plant modeling

The requirements for BE calculation have created some practices of modeling a plant with 1-D thermal hydraulics codes. In the system codes, nodes, junctions, heat structures, etc., are the components of describing the geometry of a plant. At present, pipes are typically described radially with one node, but if the diameter increases, the division into several parallel parts becomes necessary. The modeling of the phenomena in the pipes is possible with the non-homogenous two-phase model, but certain phenomena – e.g., the case of a pipe with two adjacent water flows of different temperatures, perhaps in opposite directions – cannot be simulated with 1-D nodalization. There are several flow modes in the vertical and horizontal pipes. The flow mode depends on the flow rate, interfacial friction and wall temperatures. The flow mode is important in the connections between vertical and horizontal pipes. Special models are typically needed here.

The nodalization should define at least the real boundaries between the physical parts and components in a plant. Here, isolation valves may change the boundaries of the flow system during transients. Further, special attention should be paid when choosing the boundary elevations between the thermal hydraulic code model and the neutron kinetic code model for the coupled code calculations because the hot channel and the hot rod analyses performed afterwards use the thermal hydraulic boundary conditions defined in these elevations.

NPP measurements give valuable feedback on the calculations, but one should be able to simulate the measurements – see Chapter 5.3. The data on measuring points and a detailed enough nodalization to produce the measurements are needed. Most important is to be able to calculate the significant physical phenomena correctly, even though the number of nodes is limited. Here the increasing computer calculation capability will help in the future.

A complicated flow distribution is expected in vessels. In a PWR the reactor vessel, steam generator and pressurizer are examples. The 1-D system codes may include separate models for describing the 3-D phenomena in vessel volumes and the rest can only be taken into account by a suitable nodalization. If a volume is modeled by only one vertical node chain without recirculation, a 1-D thermal hydraulics code can, e.g., enforce the flashing phenomena. In the reactor vessel and in the steam generator secondary side, parallel horizontal nodes are applied in the SMABRE models but not for the pressurizer. In the pressurizer, the large water surface may over-predict the condensation. In SMABRE, the condensation is reduced with a special node type.

The steam generator modeling has a large effect in several transient types. At VTT, the horizontal steam generator model used for VVER-440 was created in the 1980s for the RELAP code [103]. The heat transfer tubes, and the collectors in the primary side and the corresponding secondary side are divided into five levels. The vertical division is important, especially in transients where, due to a lower secondary water level, the heat transfer from the primary to secondary side is decreased. Further, internal primary side circulation due to layered tubes affects some transients [106].

In the VTT model the steam generator secondary side is horizontally divided into two parts. The outer zone, where the measurements are located, is described as a separate downcomer. This part is only a small part of the almost 11 m long and 2.3 m wide cross flow area of the steam generator. The riser part includes the heat transfer tubes described with two-sided heat structures. In SMABRE this kind of a steam generator model indicated too low collapsed water levels compared with the Loviisa plant data. With the same swell level and the same total water inventory, the collapsed water level could be increased by decreasing the void in the downcomer, simultaneously leading to a more voided riser part. This was achieved with a special junction in SMABRE to limit the carry-under flow of steam from the riser part to the downcomer within the water.

The simulation of large- and narrow-scale water level measurements is achieved at the nominal or low power states in Loviisa. During the fast depressurization transients with a strong flashing, as in the secondary breaks, the deviation of collapsed levels in the riser and downcomer may be remarkable, although the two-sided model enables strong circulation in the secondary side. The main function of the narrow-scale signal in Loviisa is the contribution to the feed water control, whereas the large-scale water level signal may cause the reactor coolant pump trip. Dense nodalizations can be used for modeling steam generators with traditional system codes [21], but the usable data to validate the horizontal mixing and carry-under flows is limited. The importance of horizontal steam generators has been underlined in a series of conferences in Lappeenranta, Finland, and the latest in Moscow, dedicated just to SGs of VVERs [107].

Several vertical steam generators have also been modeled in the SMABRE validation process, among others the once-through steam generator of TMI. Typically for vertical steam generators, the division of the secondary side into several channels was not used in the nodalization until the secondary side was actually divided by structures, as in EPR.

5.2 Flow mixing simulation

Flow is mixed due to small fluctuations and circulations, i.e. turbulence in the flow, which tends to equalize the temperature differences in the flow. The boron content of the liquid is mixed in a similar way. In addition to turbulent mixing, there is the non-physical mixing by numerical diffusion in the calculations. This tends to smooth out sharp fronts. The boron and temperature fronts are important for safety analysis in PWRs. Further, the mixing in the pressure vessel is one of the main interests in MSLB because of the asymmetric cooling in the pressure vessel. Another transient type is the inherent boron dilution [108], but the asymmetric injections to the primary circuit with and without boron should also be considered. The reasons for a boron dilution may be pure water injection or a diluted slug formed in the boiling-condensing mode in an accident or in the starting of an isolated circulation loop after maintenance. The later case concerns the VVER-440-type reactors with the main gate valves in each circulation loop. A diluted slug and a slug with higher boric acid concentration behave differently due to the density difference.

All the system codes have similar problems with the temperature and boron fronts. One solution would be the method of characteristics, but, up to now, none of the largely used system codes apply it. The thermal hydraulic solution of TRAB-3D and HEXTRAN, described in Chapter 2.2, differ from the system codes. They preserve the boron fronts in a core with adequate accuracy in normal flow conditions. The models used in, e.g., CFD codes are possible but difficult to apply here because of the sparse nodalization. Some simplified separate models are used instead, but at present different kinds of tricks are used to describe the phenomenon. The phenomenon is dependent on the detailed geometry and different practices have to be created for each reactor type. In Finland, in the boron dilution problematics, CFD calculations have been performed for Loviisa [109]. In Germany an approach of performing experiments and CFD calculations for the creation of simplified models to define the boundary conditions to a 3-D neutron kinetics code has been applied [110].

A new alternative general solution for boron fronts is under testing in APROS [111]. This second order discretization preserves better sharp gradients. For the solution of liquid enthalpy, a similar, but not general, model is available, allowing a more precise simulation of temperature stratification and propagation of a temperature front.

A 1-D node chain represents total transverse mixing in the nodalization for the system codes. At present, parallel node chains are used in several codes in order to describe incomplete mixing in the pressure vessel. In SMABRE the horizontal flow between the parallel nodes in the reactor pressure vessel before and after the core is described with cross flow junctions. The mixing here is a result of a flow between nodes driven by pressure differences. The final tuning of the mixing in SMABRE is done with the turbulent mixing model applied in the cross flow junctions. In this model the differences in the boron content and the enthalpy are reduced by the mixing between neighboring nodes.

Typically, the number of the parallel node chains – i.e. the sectors in the pressure vessel nodalization – is at least the number of circulation loops. The minimum number of sectors in the nodalizations for five reactor concepts is depicted in Figure 8. In VVERs, the hot and cold legs are situated one upon the other. Due to the hexagonal geometry of the VVER-1000 core, the division into six sectors is recommended in practice in spite of the four loops. In TMI, the different locations of the two hot and four cold legs stand for at least six sectors. The locations of the two cold and two hot legs side by side in EPR presume eight sectors until the correct boundary conditions for the 3-D core calculation in all kinds of transients are met. With this kind of method, the approximate radial effect of one separate loop on the pressure vessel flow may be achieved, although in the real plants the behavior of the flow may be more complex.

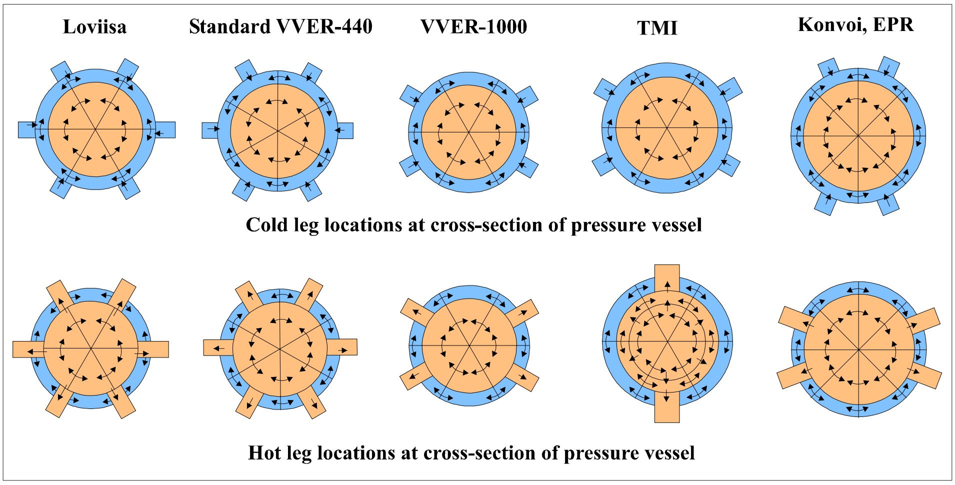

Figure 8. Simplified orientation of loop connections to pressure vessel and minimum number of pressure vessel sectors in nodalizations for five PWRs.

Flow swirling occurs in the downcomer in certain power plants. Furthermore, in several experimental tests and CFD calculations a slug coming from one loop goes through the lower plenum to the other side of the pressure vessel before the core inlet. This kind of behavior is far too special to be modeled with the 1-D system codes. On the other hand, a control rod drop in the middle of two loops in the Bohunice plant was described in Paper IV. The drop was seen to be quite similar in the two neighboring hot legs. The result indicates a quite straight vertical flow without swirling in the upper plenum.

Instead of using the small temperature differences measured in plants for defining and tuning the mixing degree, a special method using boric acid concentration in the flow is utilized with SMABRE [9]. The mixing in the pressure vessel is often expressed with the mixing coefficient describing the flow fraction not continuing in the same sector (corresponding to a certain loop) through the pressure vessel from the cold to the hot leg. For Loviisa, 20 % mixing is used according to the measurements for VVER-440 [112]. About the same mixing degree appears in the measurements in Paks NPP, but there is no swirling in the downcomer [113]. Respectively, the orientation of the six SMABRE sectors for Loviisa differs from the standard VVER-440 – see Figure 8.

Because of the non-modeled turbulent vertical mixing in system codes, the temperature diffusion vertically from node to node is also missing. The phenomenon may be important in large 1-D modeled volumes, such as tanks. The upper head in the pressure vessel of PWRs, as well as the tops of the collectors in the horizontal steam generators of VVERs, are known as 'dead ends'. In the 1-D calculation they may behave 'as other pressurizers' for the primary loops. The phenomenon is important in the safety analyses, not just because of pressure behavior but if there is no mixing, injected boric acid is mixed in the upper head and the boric acid concentration in the rest of the primary loop is overestimated. In the SMABRE code it is possible to apply the turbulent mixing model vertically as well, or to use the nodalization means described below to simulate the phenomena.

If the pressure vessel is divided into parallel node chains or sectors, the mixing is ensured if the locations of the loops are asymmetrical, but, in VVERs for example, the pressure driven mixing is missing in the upper head if the number of sectors is the same as the number of circulation loops. When the calculations are carried out with a 1-D thermal hydraulic code, the upper head temperature and boron content stay at about the initial level due to poor vertical mixing between nodes. Especially during the forced circulation, it is obvious that the temperature in the upper head should drop near to the hot leg temperature. A method with six artificial junctions from the upper head of the reactor pressure vessel to each hot leg is introduced in Paper II for VVER440. The major effect of the used mixing junctions is a more realistic upper head temperature, but they also have a smaller effect on the other main primary parameters.

5.3 Simulation of measurements

The simulation of measurements is not very relevant in the code-to-code validation because only the calculated parameters are compared. However, the simulations play an important role in the validation against plant data and in the safety analyses – e.g., the accurate simulation of measurements may have an effect on the timing of signals and, consequently, on the development of the transient in the safety analyses. Typical measurements of a plant are listed in Table 8, together with the measuring type and aspects to be taken into account in the simulation. The Table has been developed in the application and validation process of the coupled codes introduced in this thesis.

The signals of the local neutron detectors inside the core (SPND) and the signals of the thermo-couples are filtered due to the physical properties of the measurement instruments. In the SRR1 project a model was created to handle the calculated values before comparison with the measurements. The core internal SPND measurements were used in the international validation process of the coupled codes for VVERs for the first time. These measurements remarkably extend the use of real plant data for coupled code validations.

Fluid temperatures in PWRs are measured at several points and are also used in the automation for the construction of signals; the temperature is only measured at a few points in BWRs. For Loviisa, a value of 10 seconds was used for the time constant of the temperature measurement from the primary circulation loops [50], and a time constant of 30 s [Paper II] was used for the core outlet temperatures. These values have later been used as the first guess for other VVER plants, even if the time constants naturally depend on the measuring devices used.

The core outlet temperature measurements have also been used for evaluating the radial power distribution in the core. In order to succeed in the calculation the exact mass flow has to be used for the core, and, therefore, the determination of core bypasses is essential. The core bypass flow consists of several flow components, such as the flow outside the core barrel, the flow inside the barrel but outside the fuel assemblies in the BWR and VVER-440 reactors, the flow through the control assemblies and the flow in the fuel assembly central tubes. Up to now, the core bypasses for VVERs have been combined together at VTT.

The flow out through the holes in the lower part of the shrouds should be considered in order to simulate the exact 3-D power distribution in the VVER-440 reactors. Furthermore, the core outlet temperature distribution cannot be simulated in detail if the bypass flow back into the upper part of the assemblies is not taken into account. Detailed CFD studies have recently been performed on the flows in the upper part of the assemblies and the effect on core outlet temperatures [114, 115]. These features could be easier to model with the use of the internal coupling of the neutron kinetics and thermal hydraulics codes. In BWRs, the water crosses inside the fuel assemblies impose the modeling of a separate bypass for each assembly.

The readings of the core external ionization chambers should be simulated in a detailed core calculation. These readings are typically used at the plant to define the core's total neutron power, but the interdependence is not straightforward. Here the locations of the detectors are essential. Taking the vertical position of the chambers into account is important, especially when the control rods are partly in the core. A special kernel model developed in Finland to modify the HEXTRAN or TRAB-3D fluxes in core nodes [116] was used to simulate the readings of the chambers. This was applied in the Loviisa case (Paper II). Further, in Paper IV, the measurements on the opposite sides of the core are described in the case of a control rod drop in 12 seconds in a VVER-440 plant. The chambers on the opposite side do not show any change due to rod drop, because the change in the measurement is not large enough.

Generally, all the main measurements in the NPP are stepwise. The readings are updated when changes are large enough from the previous value. A large change, e.g. 25 cm in a typical step-wise measuring of the control rod position in VVER-440s, complicates the code validation.

There are several points to be considered for all the items listed in the measurements. A possible failure in the measurements should be taken into account in the safety analyses. Some measured signals treated with erroneous software or unsynchronized clocks of different measuring systems at the plant had to be considered in the validation process. Furthermore, extra filtering may be involved in all the measurements.

Table 8. Typical plant measurements and their simulation.

Measurement	Type	Aspects in simulation of measurement in transients
Fluid temperature in circulation loops of PWR	Thermo-couple	-Time constant of measuring instrument. -Effect of wall temperature.
Fluid temperature at core outlet in PWR	Thermo-couple	-Time constant of measuring instrument. -Elevation of measurement. Mixing degree with core bypass before measurement in VVER-440. -Effect of central tube in VVERs.
Fluid mass flow rate	Pressure difference measurement	-Loop mass flow produced from SG or RCP pressure difference. Not relevant in flow reversal.
Water level	Pressure difference measurement for collapsed level	-Exact vertical and horizontal locations of probes. -Density of water in measuring tubes.
	Float arrangement for swell level	-Not relevant in calculations. Some models based on local void fraction.
3D power distribution	SPND- detector inside core	-Time constants based on half-lives of isotopes in detectors. -Decreasing detector sensitivity as function of material burnup. -Lengths of detectors.
Neutron power, PWR	Core external neutron detector	-Effects of core outer structures and control rods to local neutron flux.
Position of control rods	Electromechanical devices	-Large steps in readings of VVER-440.
Void fraction	γ-rays, resistor networks	-Typically only measured in test facilities.

5.4 Safety analysis application, MSLB for Loviisa, Paper V

The safety analyses needed for the licensing of a plant in Finland are performed according to the YVL guides defining the transients to be analyzed and giving the acceptance criteria. The YVL guides are systematically updated, being more emphasized due to the new plant under construction in Finland. Applying these requirements to the operating plants and to a plant under construction is somewhat different. The YVL guide 2.2 [117] concerns the safety analyses and categorizes the anticipated transients, postulated accidents and severe accidents, and sets boundaries as probabilities for each category. At present, the Finnish regulations include more precise requirements in several points than is the case in many other European countries [118]. Categorizing ATWS transients to the postulated accidents is an example of this feature.

The YVL guide gives several detailed regulations for the transients to be analyzed. The initial state before a transient should be the nominal state, or another state if the consequences are more severe. This naturally presumes that they are known to be more severe. In many transients the nominal state or a standby state is typically considered to lead to the most severe result. Further, the number of redundant systems available is reduced. The single failure criterion is applied, but, in addition to one broken component or system, one more component or system is presumed to be unavailable due to maintenance or testing. Further, the YVL Guides list several conservative points to be taken into account in models and the models of plants. In practice, the main part of the conservative points is presumed before the analyses because the number of cases analyzed with the coupled codes is limited. The present acceptance criterion concerning MSLB is that no recriticality is allowed in the design basis accidents (DBA).

As an example of a safety analysis, the MSLB calculations for the Loviisa NPP performed in 1996/97 are described in Paper V. The analyses were performed with HEXTRAN-SMABRE as part of the plant modernization and power upgrading project. Only the nominal state was considered as the initial state in the project because the low power conditions were not affected by the nominal power increase.

This analysis is a good example of the benefits of the parallel coupling of codes. The SMABRE code was used with point kinetics for extensive scoping calculations and the variations with the highest reactivity potential were analyzed using the coupled code HEXTRAN-SMABRE. In this quite loose coupling the input data defined for both codes are quite similar in the stand-alone and coupled code calculations. Generally, the results are similar too, until the differences between 3-D kinetics and point kinetics begin to appear. With the fast running code these preceding studies could be done with fairly low computer capacity.

MSLB is somewhat different for each plant, but in all PWRs the potential hazard is the recriticality due to the strong core cool-down and, consequently, the possible fuel damage. The cooling in the core is strongly asymmetric if several RCPs are in operation. Here the accurate plant modeling is a demanding task because various complicated processes, such as asymmetric power generation and mixing in the reactor vessel, as well as various protection and conventional automation signals, contribute to the intricacy of the scenario. The plant model for SMABRE was briefly described in Chapter 3.3, where the nodalization scheme of the primary loop for SMABRE is shown in Figure 3. The quite weak mixing in the VVER-440 pressure vessel was discussed in the context of flow mixing. In this project the secondary side was remarkably extended. The nodalization scheme of the secondary system is shown in Figure 9, where various break locations assumed in the analyses are shown. The pipelines, which can empty the water content into the broken steam generator in one of the calculation cases, are marked with the blue color.

The scoping studies included various assumptions on the break size, the operation of the reactor coolant pumps and the performance of other systems. In order to make a realistic prediction of the plant behavior, care was taken to model several critical details, such as the level measurements in the horizontal steam generators, the feed water system and turbine control.

The best estimate analysis offers the possibility of evaluating the plant safety in a more realistic manner and helps to eliminate the excessive conservatism, which may even mask the real problems. Here, in all the cases studied, the complete shutdown of the reactor was assumed upon reactor scram. In order to stay on the safe side, some conservatism was imposed on core reactivity characteristics – e.g., a stuck control rod was assumed. The sensitivity analyses were not performed for these analyses as the scoping studies gave more confidence in the results.

The calculations showed that only a mild return to power after the reactor scram is to be expected under the worst conditions in MSLB at Loviisa.

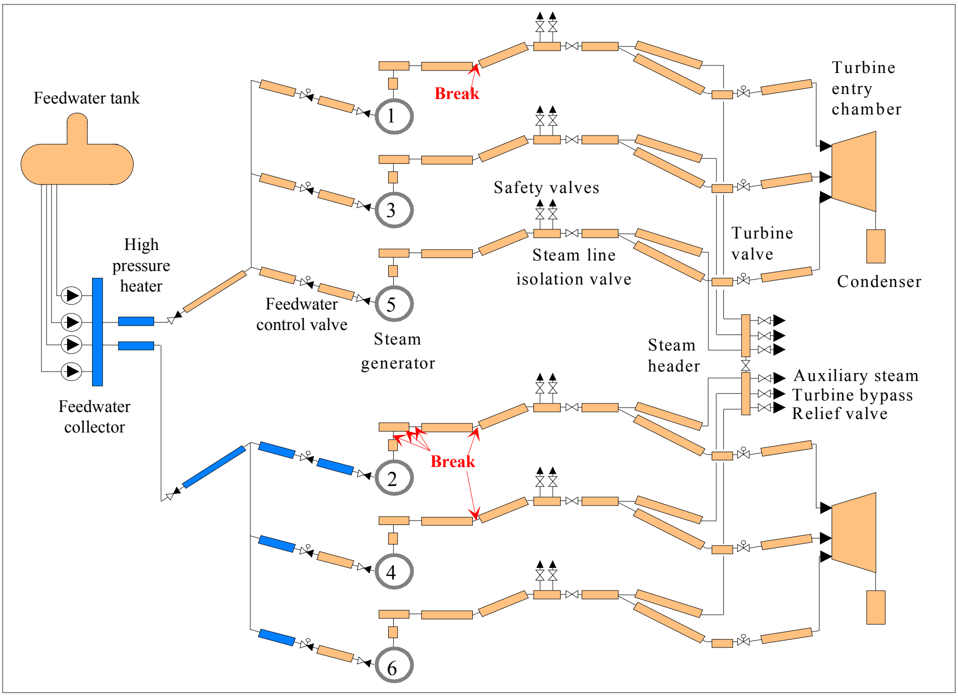

Figure 9. Secondary system of the Loviisa VVER-440 plant described in SMABRE nodalization for the main steam line break cases.

5.5 Comparison of MSLB analyses

The MSLB for the standard VVER-440 plant has also been calculated in the 6th AER benchmark, described in Chapter 3.3.2. The HEXTRAN-SMABRE results for MSLB at the nominal state are quite similar in the two cases, despite the differences between the plants.

The MSLB analyses made for the Loviisa NPP were a basis for the models used at VTT in the AER 6th benchmark problem. Still, the results of the Loviisa analyses and the AER benchmark cannot be directly compared. In addition to the slightly different assumptions, the low primary pressure (10 MPa) causes the coast down of all the Loviisa main coolant pumps in this kind of double-ended steam line break (200 %). Alarge spectrum of calculations was performed (50 % – 400 % break size) for Loviisa, and, as a result, it was stated that the worst case might be reached with a smaller break size.

When comparing the VTT variations in the AER 6th benchmark with the variations calculated for TMI-I, Chapter 3.3.1, it can be outlined that due to the effectiveness of steam to carry water out through the break the results are more severe in the TMI plant with the vertical OTSG. This feature is clearly less severe in VVER-440 due to the earlier steam generator tube uncovering, which ends the cooling of the primary water. Generally, energy is taken out of the system more effectively with the steam-phase break flow and the pressure reduction is larger. The mass flow out is bigger with the water-phase break flow and the steam generator water inventory decreases faster.

The different break flow models of the system codes do not remarkably differ from each other. The Moody model is used in SMABRE, which does not presume as detailed modeling of the geometry near the break as many other models. Huge steam velocities are met in the steam lines after the large break opening and all the water in the steam lines is probably flowing out with the steam. The behavior of the separators and dryers of steam generators is most indistinct under these kinds of extreme conditions with huge flow velocities and strong flashing. These components should be modeled in more detail in the system codes. Another possibility is to assume a more conservative flow mode at the steam generator outlet in safety analyses. Further, within the context of the AER 6th benchmark, the five straight vertical pipes from SG to steam lines in VVER-440 were found to be important for phase separation.

The final cooling of the primary water is a function of the break flow mode and several other details, such as the steam generator geometry involved, mixing in the pressure vessel, systems operating during the MSLB transient – e.g., available feed water to the broken steam generator – and the speed of the isolation of broken steam generator. A smaller break size with a weaker cooling effect may, however, lead to lower primary temperatures due to the continuously decreasing pressure in the broken steam generator. That is why the biggest break size does not automatically lead to the most severe result. Definitely, the final possibility of recriticality depends on how the boric acid concentration in the primary water can eliminate the reactivity effect of the core overcooling.

Stopping the cooling down of primary water is the first target in the prevention of recriticality in all kinds of plants. At this point the prevention of the feed water injection into the broken SG as well as the prevention of heat transfer in the broken SG by stopping the main coolant pumps are favorable operations in cooldown cases. The second target is the early boration of the primary water.

6. Coupling of thermal hydraulic and fuel rod codes, Paper VI

The aim of this chapter is to introduce a novel approach to improving the special fuel rod modeling in transients. The author realized the coupling between the independent codes GENFLO and FRAPTRAN, excluding the internal modifications needed in FRAPTRAN, and prepared the test cases for the coupled code.

Several validation efforts have indicated that the fuel rod models need improvements to the reactor dynamics codes. This conclusion has been based on relatively mild transients. The accuracy of these models in some boron dilution and control rod ejection cases may be even more important. More detailed fuel rod simulations in the reactor dynamic codes may also be achieved by coupling the codes to actual fuel codes. However, the aim of the coupling of codes in this chapter and Paper VI is the expectation of better thermal hydraulics in a special fuel transient code.

The two-phase thermal hydraulic simulation capability is needed in several codes. The traditional thermal hydraulic system codes are often too heavy for these purposes, e.g. in the couplings of several codes or, especially, in the simulator applications where the computing time is relevant. The traditional thermal hydraulic codes and dynamic changes of fluid volumes are not possible. This feature is especially needed in severe accident applications, where fluid volumes may change due to the melting of solid structures.

The thermal hydraulic model GENFLO [10] has been developed at VTT for special applications of the coupled codes in close connection with the SMABRE code. Hence, as in SMABRE, the conservation equations in GENFLO are solved for the phase masses, mixture momentum and phase energies. The phase separation is solved with the drift flux model. Besides, the solution method with the fast running feature is similar to SMABRE. However, the simulation capabilities have here been extended to high temperatures consisting of the cladding oxidation and a model for quenching fronts. Further, the possibility for dynamic changes of fluid volumes has been introduced by a parameter of node porosity. In spite of the similarities to SMABRE, GENFLO is not a system code but rather a thermal hydraulic model for special uses for phenomena inside the pressure vessel.

The initiative for the GENFLO development came in the simulation of the thermal hydraulics of a BWR pressure vessel in a recriticality analysis. GENFLO was coupled with the two-dimensional transient neutronics code TWODIM [119]; the two codes together are known as RECRIT. The recriticality with one rapid prompt peak was expected during a severe accident scenario, where the control rods had already melted and the emergency core cooling system was started. In this phase GENFLO was validated against large break LOCA reflooding experiments and in severe accident conditions against the QUENCH tests.

The second application of GENFLO [120] was APROS-SA (Severe Accident), where several separate codes were coupled. APROS-SA is part of the APROS simulator developed for operator training and capable of real time simulation. Here the core heat-up and oxidation, metal and fuel pellet relocation, and corium pool formation on the upper tie plate and in the lower plenum are calculated. In this application the GENFLO model simulates the PWR pressure vessel thermal hydraulics. The simulation capability in APROS-SA covers the phenomena met in the VVER-440 reactors and, to date, the model for Loviisa exists and the model for some other VVER-440 is under preparation.

Paper VI presents the third application of GENFLO, the coupling with the U.S. Nuclear Regulatory Commission’s (USNRC) updated version of the FRAPTRAN fuel transient code [121]. The need for the best estimate analysis with the fuel transient codes has grown together with the requirements to analyze ATWS in the basic licensing calculations. Further, the higher fuel burnups set requirements for a more detailed simulation of the local phenomena in the fuel rod. GENFLO replaces the fairly simple thermal hydraulics of the fuel transient behavior code in this process.

The codes and the main principles of the coupling are described in Paper VI. The parallel way of the coupling offers flexibility in exchanging the necessary data between the codes, although the solution methods of the codes differ significantly. This quite loose coupling has also proved to work here.

In the analysis, GENFLO simulates the subchannel around a single fuel rod and delivers the heat transfer on the cladding surface to FRAPTRAN. The FRAPTRAN routines are then used to yield the temperatures of the fuel rod and the deformation of fuel pellets and cladding, including potential ballooning. The stand-alone FRAPTRAN-GENFLO without a system code has been successfully applied in the analyses of the LOCA experiments in the HALDEN research reactor with real fuel [122].

The first power plant application of the coupled code was the case of a large break loss-of-coolant-accident (LBLOCA) in the Loviisa VVER-440 plant. The system behavior was calculated with the APROS code. Simultaneously, the boundary conditions for the FRAPTRAN-GENFLO hot rod/subchannel analyses were defined in the hot channel model of APROS. No cladding ballooning was predicted with the coupled code in the FRAPTRAN-GENFLO calculation. The separate FRAPTRAN calculation would suggest ballooning, cladding deformation and rod failure due to the partly unrealistic boundary conditions.

The second power plant case studied an actual instability incident in a BWR reactor. Here the system behavior was calculated with the 3-D neutron dynamics code TRAB-3D. The real event was extended by assuming a hypothetical failure of the reactor trip leading to continued oscillations. The fuel rod was assumed to have a high burnup. Calculation of this instability case with FRAPTRAN-GENFLO showed that the subchannel around the hot rod was wetted and no sudden rod failures occurred until flow reversal – see Figures 9, 12 and 13 in Paper VI. The gas gap was expected to be closed at the end of the calculation.

The cases here should be considered more as examples, and the final safety analyses would need a more carefully studied input and boundary conditions for both cases. However, the results of this first version of the coupled FRAPTRAN-GENFLO code were most encouraging and bode well for the further validation against experimental data. The experiences with the GENFLO code have pointed out that the efforts needed for the code development and validation can be minimized with this kind of flexible model for several purposes.

7. Sensitivity and uncertainty analyses, Paper VII

This chapter introduces sensitivity and uncertainty analyses, applied in transient analyses for the first time in Finland. The author had main role in carrying out the analysis in Finland. The statistical part of the analysis was performed at GRS.

The progress to a more detailed simulation inside the fuel assemblies, the pin power reconstruction, and better fuel rod models, as well as the progress towards a detailed 3-D simulation of thermal hydraulics, are the present trends in the reactor dynamics code development. In performing safety analyses, these trends may be completed by adopting the sensitivity and uncertainty analysis as a routine practice.

In the traditional safety analyses the required conservatism has mainly covered the uncertainties of the results and their sensitivities to input parameters. Obviously, the sources for uncertainty are numerous when all the physical and technical reasons are taken into account – engineering factors, flow turbulence, approximate code models, simplified plant models, code user effects, and so on [123].

The 3-D behavior, especially in the core, creates a situation where it is not always possible to predict the conservatism of the final result from the input parameters. As a result of the comparisons of analyses performed with 1-D and 3-D neutron kinetics, 3-D calculated results are typically less severe, but not always [41]. Obviously, the conservatism may become obscure due to more and more complicated and more and more detailed systems and codes [124]. These are some reasons for the BE calculations. On the other hand, the authorities still request conservative safety analyses. This may be fulfilled by performing sensitivity and uncertainty analyses (UA) [125].

Several institutes in the nuclear field have created their own methods for performing UA, often relying on their own code system. As part of the VALCO project, the GRS method was applied for two transients (Paper VII), 'Drop of one turbine to house load level experiment at the Loviisa-1' and 'Turn-off of one from two working steam generator feed water pumps at the Balakovo-4', both already calculated in the previous EU project SRR1 (Papers II and III).

The applied GRS method consists of selecting and identifying independent input parameters relevant to uncertainties and defining ranges and a probability distribution for them. The uncertainties of output values are estimated by varying all these parameters at the same time. The sets of values for the selected parameters are determined randomly using the Monte Carlo method. The minimum number of sets and the number of runs is defined by Wilk's formulae [126] in order to determine the tolerance limits. In this method the number of runs does not depend on the number of varied parameters. In the Loviisa case, 10 independent input parameters were selected. In order to obtain the defined statistical tolerance limits of the output values, the total number of 100 runs was chosen. After performing the 100 calculations, the output values were evaluated with the statistical code package SUSA at GRS.

The primary pressures from 100 runs are presented in Figure 10. Here, only a few runs clearly deviated from the others, and the same was observed by the other participants in their calculations. In the Balakovo case, small deviations of the input parameters led to a totally different transient scenario in several participants' results. This is an example of why a more realistic and more completed model may be needed for UA than for the original calculation.

As an example, Figure 11 shows the sensitivity of the primary pressure on several uncertain parameters in the Loviisa transient. The sensitivity is expressed with the rank correlation coefficient of the parameters during the transient. The relative importance of the parameter may be seen dynamically, but the rank correlation coefficients with absolute values below 0.2 are not significant. The sensitivity in the initial state compared with the transient was overrated in this case. This was basically a result of small differences in the assumption of the same initial state. Here, the parameters affecting the fission power – i.e. the control rod insertion (2 in Figure 11) and the reactivity worth of the control rod (3) – had the largest effects on the primary pressure. The other parameters having a significant effect were the upper head mixing flow (5), the operation of the make-up pumps (6) and the pressurizer heater power (8). This is reasonable and foreseeable.

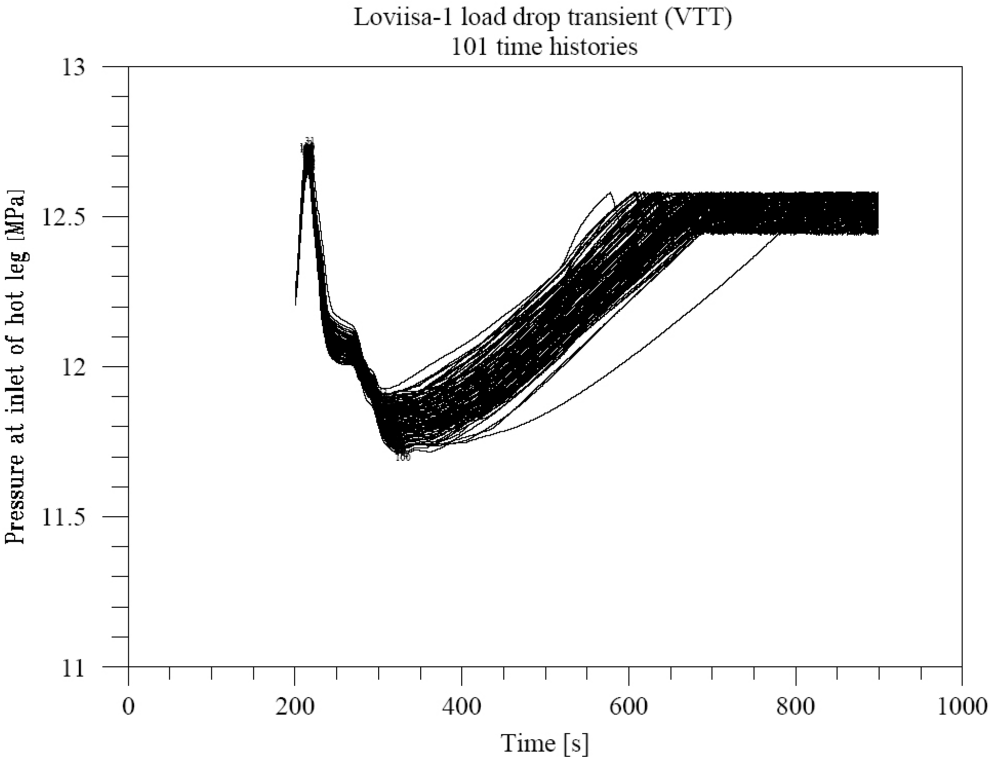

Figure 10. Primary pressure in 'Drop of one turbine to house load level experiment at the Loviisa-1' calculated with HEXTRAN-SMABRE in the original case and 100 UA calculations [127].

A comparison between the results of different participants can be made in a UA with several participants. The effect of the heat transfer coefficient in the gas gap on the power level was not as remarkable in the HEXTRAN-SMABRE calculation as in the other calculations (Paper VII). The reason may be the more realistic temperature-dependent gas gap model in HEXTRAN, whereas the constant value models were used in the other calculations.

The UA methods are not perhaps an easier or faster way to reach confidence in the conservatism of results. As in the traditional safety analyses, similar difficulties are met and expertise is needed. The conservatism of each coefficient has to be thought through in detail in the traditional safety analysis, and all the selections jointly affect the result. This may also contribute to the over-conservatism. In applying UA, the choice of uncertainty parameters, and their distributions and ranges, is essential. Typically, the codes need so many independent input parameters that analyzing all of them is difficult.

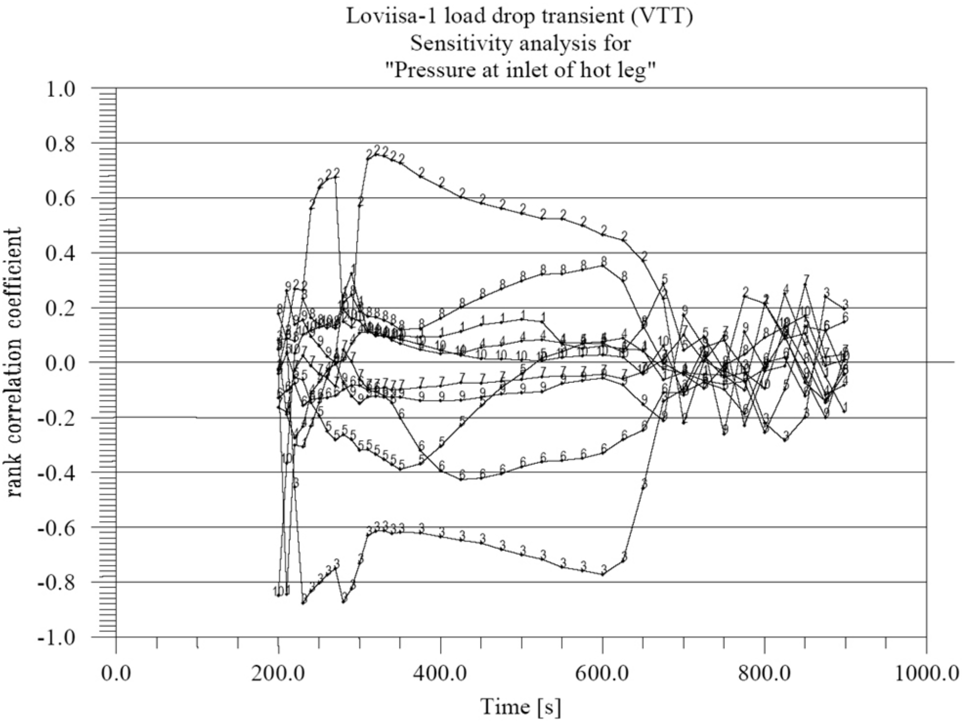

Figure 11. Sensitivity analysis for primary pressure based on the 101 runs with HEXTRAN-SMABRE for the 'Drop of one turbine to house load level experiment at the Loviisa-1'[127].

In this case UA was performed for the transient and, basically, the coupled code uncertainties were studied. It did not include the uncertainties considered in hot channel and hot rod analyses, but the results could be used to define the uncertainties originating from the transient itself. Furthermore, the methods could be exploited to fulfill the requested statistical approach included in the present Finnish acceptance criteria on the number of fuel rods undergoing heat transfer crises in transients.

At present, plant safety is considered in parallel with two approaches: deterministic safety analyses and Probabilistic Safety Assessment (PSA). These two approaches co-operate in some sense. Now, UA creates a closer link between them; the UA method can, e.g., produce useful distributions to be applied in PSA.

8. Conclusions

This dissertation focuses on the validation and application of the coupled codes developed in Finland for the safety analyses of the light water reactors in design basis accidents. The validation efforts and applications of the thermal hydraulics code SMABRE and the three-dimensional neutron kinetics codes are gathered up both for the separate and coupled codes. The parallel coupling method has proven to be reliable in the neutron kinetics / thermal hydraulic code couplings. Generally, the thermal hydraulic phenomena are well simulated with the one-dimensional code SMABRE, and methods for handling three-dimensional thermal hydraulic phenomena are given in the thesis. The code development and coupling of codes for the safety analyses is discussed and the present situation and possible future directions of the codes involved are described.

Data for the code validation consist of the experimental data from test facilities, numerical benchmarks and data measured in real nuclear power plants. The nuclear core is relevant for the couplings of neutron kinetics and thermal hydraulics codes. Data on ten real NPP transients was collected and reported for the code validation in the European Union projects. These projects showed that the coupled neutron kinetic / thermal hydraulic codes under consideration are capable of simulating the analyzed VVER plant transients. Hence the work contributes to increasing confidence in the results of the code systems, but several improvements to the codes are recommended. Here, the more accurate fuel and control rod models are examples. The international OECD/NEA and AER benchmarks have also been useful forums for the code-to-code validation of the coupled codes.

In order to apply a code system to the safety analyses, the validation against real plant data is highly recommended, preferably against each plant type involved. All the code updates and models should be validated against the whole validation material in the code development. In Finland, with the relatively large and successful code development in this field, such a validation should not be belittled or forgotten. On the other hand, all the operating plants are continuously measuring the main plant parameters and creating data for the code validation during plant transients, even if the data transformation to a suitable format needs some effort. This presumes a certain ambition and expertise on the part of the power companies too.

However, the real plant data has its problems in the validation. The correct interpretation of plant measurements is important. The unmeasured mass flows, the non-specified core bypasses, and the deviations between real and reported set points of operating systems are typical difficulties in utilizing inevitably deficient real plant data. To a certain extent, this supports the code-to-code validation. The plant modeling aspects, and the simulation of plant measurements emerging in the applications and validation efforts, are gathered in this thesis.

A main steam line break in different kinds of plants is dealt with in this thesis, and is also used as an example of the application of the coupled codes in the safety analysis. The potential hazard in the main steam line break is the recriticality due to the core overcooling. The maximum overcooling is a result of several details, but a smaller break size with a weaker cooling effect may, however, lead to lower primary temperatures due to the continuously decreasing pressure in the broken steam generator. This is why the biggest break size does not automatically lead to the conservative result.

As a concurrent approach to developing the fuel rod models in the dynamic codes, a new coupling of a thermal hydraulic code with a fuel transient performance code is described in this thesis. The loose coupling has successfully been realized for the improvement of thermal hydraulics in a fuel transient behavior code.

As the codes, systems and transients become more complicated, the conservatism of the final results in safety analyses is not clear. Hence the sensitivity and uncertainty analysis is introduced, a new approach and a recommendable future practice in the Finnish transient analyses. This has been tested in part of an EU project utilizing the Loviisa plant data. Further, the sensitivity and uncertainty analyses create a closer link between the traditional transient analyses and the probabilistic safety assessment.

The basics for the thermal hydraulics systems codes were created tens of years ago, but there is still need for further code development. Accurate models play a key role in reliable safety analysis, where unexpected features should also be simulated. Several challenges exist in the code development when the calculation mesh becomes denser and the best estimate calculation is requested in the modern safety analysis. The nodal neutron kinetics models expect a lot from the thermal hydraulics in the code couplings, and, even today, the uncertainties in the thermal hydraulics are clearly larger than in the neutron kinetics. Expanding the 3-D two-phase simulation capability to thermal hydraulics is one of the challenges for the foreseeable future. Here the modeling of the large pipes and vessels, such as the horizontal steam generators and the open reactor cores without the fuel assembly shrouds, greatly benefits from 3-D thermal hydraulics.

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Appendices I, IV, V and VII of this publication are not included in the PDF version. Please order the printed version to get the complete publication (http://www.vtt.fi/inf/pdf/)

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In: Annals of Nuclear Energy 2002.
Vol. 29, No. 1, pp. 255–269.

annals of
NUCLEAR ENERGY

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Annals of Nuclear Energy 29 (2002) 255–269

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Validation of coupled neutron kinetic/thermal-hydraulic codes. Part 2: Analysis of a VVER-440 transient (Loviisa-1)

A. Hämäläinen [a,*], R. Kyrki-Rajamäki [a], S. Mittag [b], S. Kliem [b], F.P. Weiss [b], S. Langenbuch [c], S. Danilin [d], J. Hadek [e], G. Hegyi [f]

[a] *VTT Energy/Nuclear Energy, PO Box 1604, FIN-02044 VTT, Finland*
[b] *FZR Forschungszentrum Rossendorf, PO Box 510119, D-01314 Dresden, Germany*
[c] *GRS, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH Forschungsgelände, D-85748 Garching, Germany*
[d] *KI- Russian Research Center "Kurchatov Institute", Nuclear Reactors Institute, Kurchatov Square 1, 123182 Moscow, Russia*
[e] *NRI Nuclear Research Institute Rez plc, 250 68 Rez, Czech Republic*
[f] *AEKI- KFKI Atomic Energy Research Institute, PO Box 49, H-1525 Budapest, Hungary*

Received 12 February 2001; accepted 28 March 2001

Abstract

Several three-dimensional hexagonal reactor dynamic codes have been developed for VVER type reactors and coupled with different thermal-hydraulic system codes. Under the auspices of the European Union's Phare programme these codes have been validated against real plant transients by the participants from 7 countries. Two of the collected five transients were chosen for validation of the codes. Part 1 of this article consists of validation against VVER-1000 reactor data. This second part is focussed to validation against measured data of 'One turbo-generator load drop experiment' at the Loviisa-1 VVER-440 reactor. The experiment was performed just after plant modernisation and more measured data was available to validation than in normal operation of real plants. Good accuracy of the results was generally achieved comparable to the measurement accuracy. The confidence in the results of the different code systems has increased, and consequences of certain model changes could be evaluated. © 2001 Elsevier Science Ltd. All rights reserved.

* Corresponding author. Tel.: +358-9-456-5023; fax: +358-9-456-5000.
E-mail address: anitta.hamalainen@vtt.fi (A. Hämäläinen).

PII: S0306-4549(01)00039-1

1. Introduction

In the SRR1/95 Phare project of the European Union, suitable measured VVER transient data have been collected and applied for the validation of coupled codes. The validation against real plant data includes the whole neutronic calculation system, i.e. nuclear data, burnup calculations, and reactor kinetics. Earlier the three-dimensional neutron kinetic codes including their nuclear data bases, and the thermal-hydraulic system codes have been separately validated against experiments. The coupling of these codes should be validated against real plant data.

Five well-documented transients have been made available in the Phare SRR1/95 project.[1] The transients as well the validation against VVER-1000 measurements are introduced in part 1 (Mittag et al., 2001) of this article and this part 2 reports about the validation against VVER-440 Loviisa-1 measurements. The test reports of the Loviisa-1 experiment prepared by the power company and some interpretative documents were delivered to all participants. Thus, the comparison between calculation and the measurements could be made by each participant. However, due to a small amount of measurements and inadequacy of modelling the plant, the result is somewhat dependent on the way each participant has decoded the real plant data.

2. Codes

The coupled neutron kinetic/thermal-hydraulic codes listed below have been validated against real VVER-440 transients. Each participant had to use his/her own nuclear data. The nuclear data libraries applied in the calculations have been generated by the codes given on the right-hand side of the list below. The coupled codes and the methods of coupling are described in more detail in the given references and in part 1 of this article.

• VTT	HEXTRAN/SMABRE (Kyrki-Rajamäki, 1995; Miettinen et al., 2000),	CASMO-HEX
• KI	BIPR8/ATHLET (Lizorkin et al., 1992),	KASSETA
• FZR	DYN3D/ATHLET (Grundmann et al., 1995; Teschendorff et al., 1996),	KASSETA
• NRI	DYN3D/ATHLET,	KASSETA
• AEKI	KIKO3D/ATHLET (Hegyi et al., 1998),	KARATE

[1] The Phare Programme is a European Union initiative which supports the development of democratic nations within a prosperous and stable Europe. Phare does this by providing grant finance to support the process of economic transformation and to strengthen newly created democratic societies. The aim of the SRR1/95 project was to improve the verification of coupled neutron-kinetic/thermal-hydraulic codes in Czech Republic, Bulgaria, Hungary, and Slovak Republic. The facts, opinions and views of this article are the authors' and not necessarily the Commission's.

3. Loviisa-1 VVER-440 transient

The experiment was carried out 12 November 1997 on Loviisa Unit 1. The reactor power had been increased by 9.1% to 1500 MW two weeks earlier.

The transient was initiated by load drop of one turbo-generator, i.e. the electric power output of the plant was suddenly reduced by half. At the moment of the generator drop, the nuclear power production in the reactor core was at 100%. Shortly after transient initiation, the reactor control system started to reduce the reactor power by inserting the control rod group number six. When reactor power was 84%, the reactor power control system was erroneously switched off. Therefore, the operator was forced to take care of the power control onwards from that point of time. The neutron power was reduced down to 60% within some 100 s.

As a result of power reduction, the hot leg temperatures of the primary circuit decreased. Moreover, the cooling of the primary circuit through the steam generator was reduced because of increasing steam pressure at the secondary side. Therefore, the cold leg temperature first increased significantly. Some 20 s later this temperature also started decreasing. Also the primary circuit pressure first increased, but was quickly reduced by spraying in the pressurizer. Later on, the reducing nuclear power decreased the pressure, so that the pressurizer heaters were switched on to stabilize pressure at its nominal level. On the secondary side, pressure also started increasing sharply, but was quickly brought back to normal by opening the turbine bypass valves, before the decreasing nuclear power took effect here, too.

For validation of the codes, the number of used measurements is fairly important. Especially during the normal plant operation, the number of measurements applicable for validation is not numerous compared to experimental facilities, and some essential variables like mass flow measurements are missing. In the Loviisa experiment the accurate events were recorded, but only in the early phase of the experiment (about 30 s). Besides, due to manual control rod movement the accurate control rod positions are not fully known.

4. Burnup and steady-state calculations

To obtain the reactor state before the transient, the burnup state at 71 full-power days of the 21st cycle of Loviisa-1 was calculated by all participants, based on the given burnup distribution from the beginning of the 19th cycle, using the given operation history. All the calculations were performed in the 1/6 symmetry of the core. The axial node lengths were 25 cm each, except the lowest and highest ones amounting to 22 cm. All the comparisons are made in this axial nodal geometry, however, in some calculations the node heights were slightly different.

The comparison of the radial distribution of the assembly average burn-up, at the initial state of the transient, is given in Fig. 1. The reference burn-up distribution was created with HEXBU-3D using the nuclear data generated by CASMO-HEX, which are used in the Loviisa fuel management calculations. The calculated burn-up

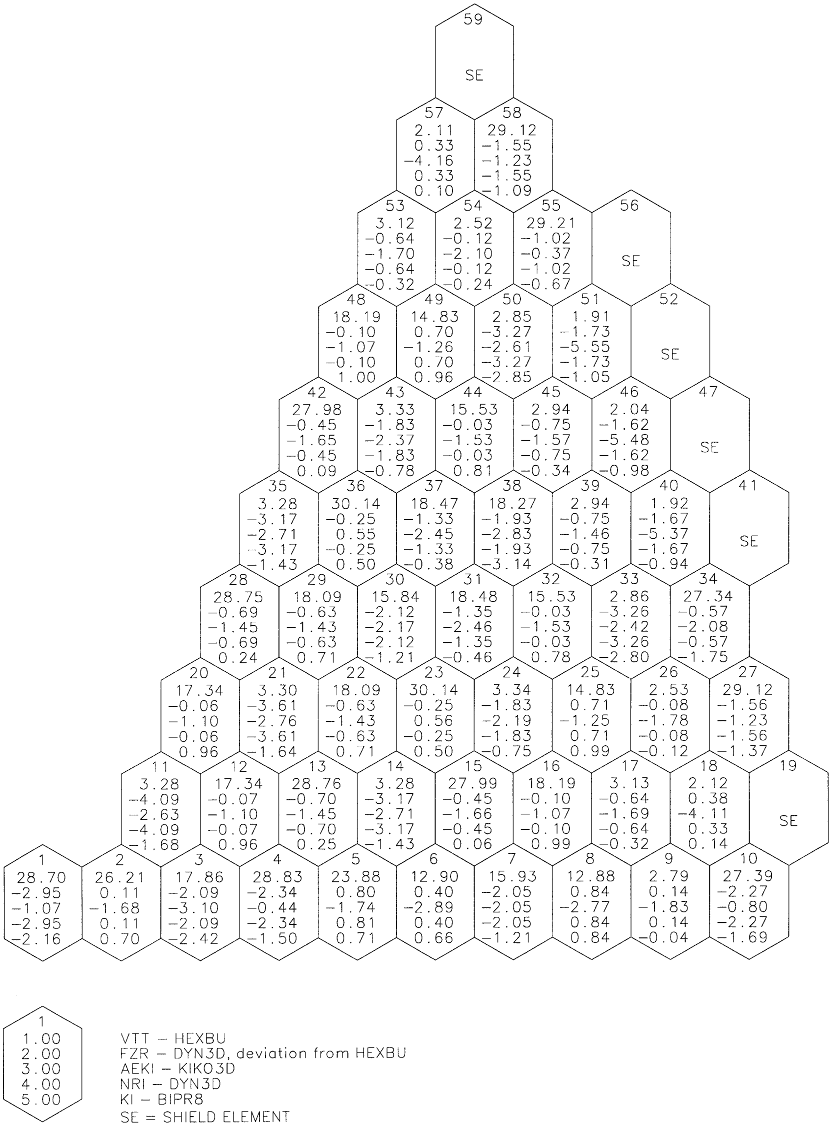

Fig. 1. Assembly average burn-up (MWd/tnU) and deviations (%) from the measurements/HEXBU.

distributions are close to each other, the maximum deviations are less than 5%. Both DYN3D/KASSETA results are naturally identical.

The measured critical boric acid concentration was 5.53 g/kg and it was reproduced by VTT with the accuracy of 1.3%, FZR with 2.2%, AEKI with 1.5%, NRI with 2.3%, and KI with 6.1%.

In VVER-440 reactors, there are fuel assembly shroud tubes preventing the coolant mixing between the assemblies. Hence, core outlet coolant temperature measurements can be used for a radial power distribution validation. Fig. 2 shows the average of the measured outlet temperatures and deviations of the calculated values from them (only calculated values where no measurements are available) in the stationary state prior to the transient. The maximum deviations are about 1.5°, but mostly the differences are well below 1°, which corresponds to a relative power difference of about 3%.

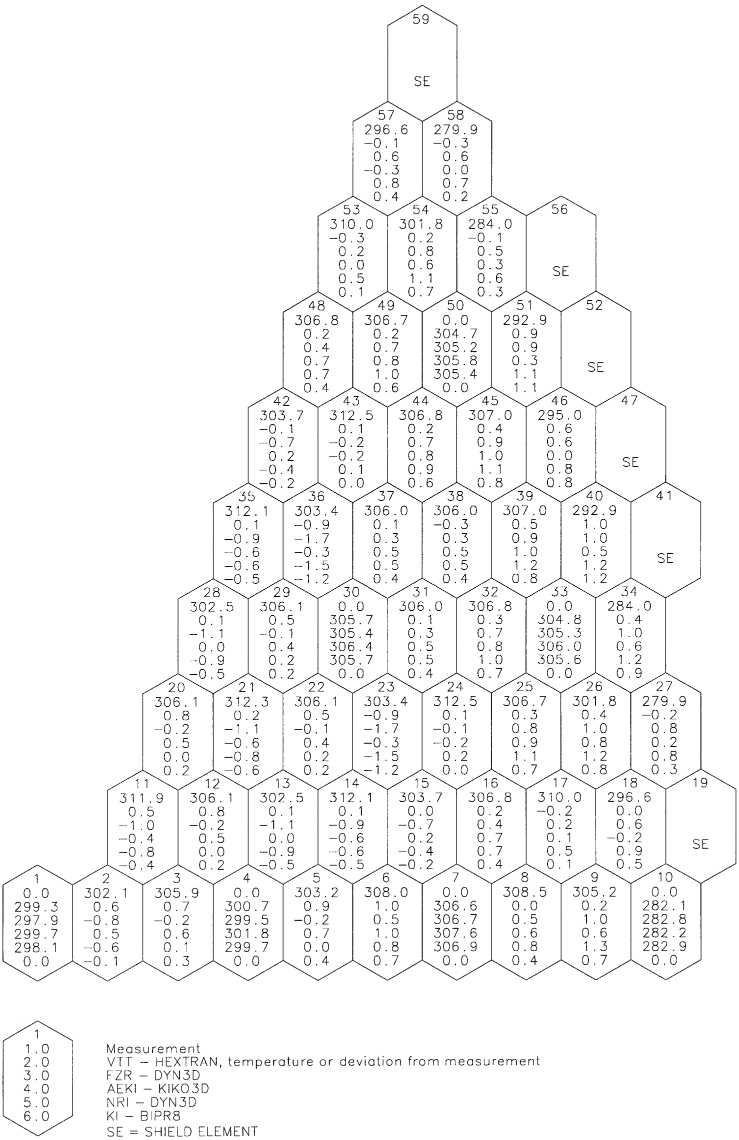

Fig. 2. Average of measured assembly outlet temperatures and deviations of calculations (°C) from measurements (assemblies 1, 4, 7, 10, 30, 33, 50: no measurements — only calculated temperatures).

5. Transient calculations

5.1. Secondary side

The turbine trip at the beginning of the experiment leads rapidly to a steam header pressure increase and the turbine bypass valve opening. In all ATHLET calculations, the measured steam header pressure was used as a boundary condition (Fig. 3). In the SMABRE (VTT) input deck, the secondary system was modelled in detail. Only the secondary-circuit valve set points were slightly modified in order to reproduce the cold leg temperatures according to the measurement. This consists of the assumption that there is an, at least partly closed, valve in the middle of the steam header. This leads to two differing steam header pressures at the halves of the steam header connected to the two turbines.

The measured mass flows of feed water pumps and positions of the feedwater valves in the feedwater lines of each steam generator were provided in the test report, but the temperature of the feedwater was not known. The average of water levels in the steam generators (Fig. 4) is very well described by VTT and KI. In the other calculations, the agreement with the measurement is worse. However, it is not important for the transient evolution because the steam generator heater tubes are always covered by water and the heat transfer from primary to secondary side is evaluated correctly.

Only a few comparisons with the measurements of the secondary side parameters are shown here because they were mostly used as input parameters in the ATHLET model, concentrating on the calculation of the primary side and the validation of the core coupling. The results of the more detailed SMABRE model agreed well with these measurements.

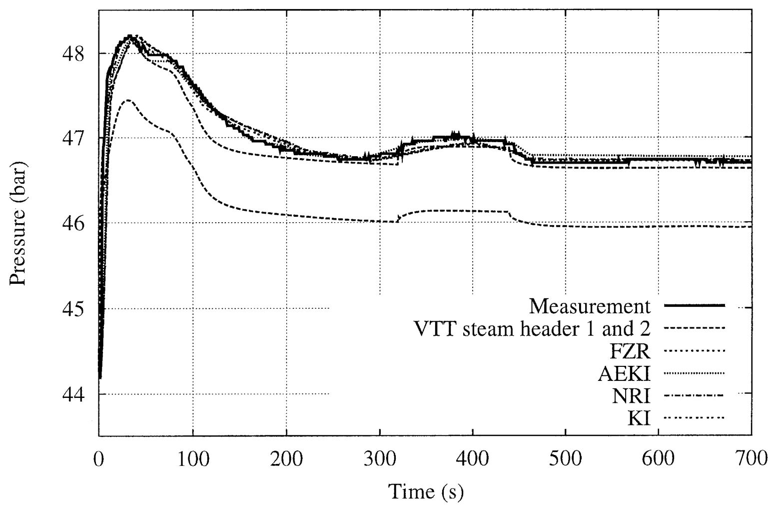

Fig. 3. Measured and calculated steam header pressures.

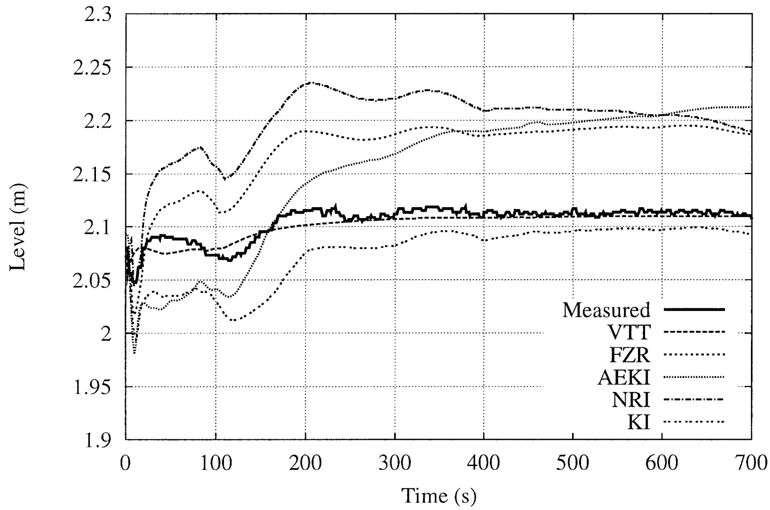

Fig. 4. Measured and calculated average collapsed water level in six steam generators.

5.2. Primary side

The temperature peak in the first phase of the transient (Fig. 5), which is caused by the secondary-side pressure peak, is very well reproduced by all calculations. After some 200 s, the temperatures calculated by VTT and KI are in best agreement with the measured values. The reason for the good VTT result is the tuning of the steam-header valve position, mentioned above. The lower cold leg temperature of

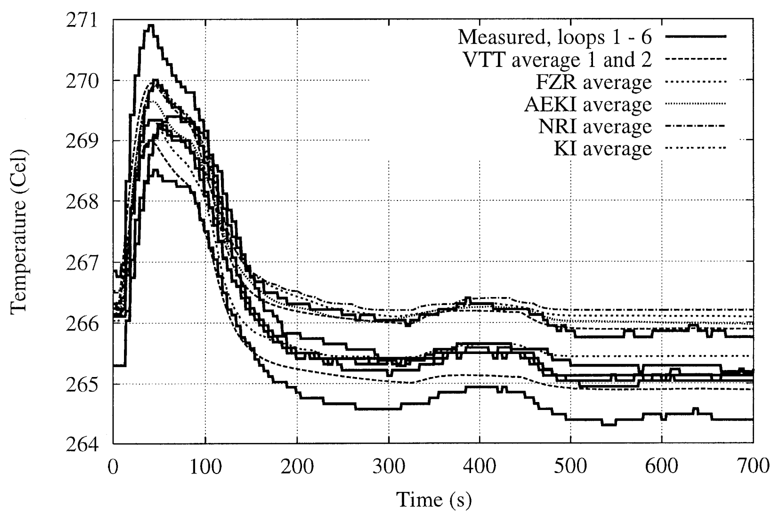

Fig. 5. Measured and calculated temperatures in six cold legs.

KI among the ATHLET users results from the bigger mass flow rate. However, all the calculated cold leg temperatures do not differ more than 1° from the average measured curve, and are almost inside the range of the measured values of the six different loops. The wide range of the measurement values in different loops is well shown in Fig. 5.

Fig. 6 shows that the following increase of primary pressure is quite well predicted in all calculations. This leads to a pressurizer heater switch off, letdown opening and the pressurizer spray valve opening. The subsequent decreasing of the pressure and the later pressure increase caused by the pressurizer heaters are also rather well reproduced by the codes, although some assumptions about the operation of the primary circuit systems were different in the calculations. In nearly all the calculations, the operation of pressurizer spray and heaters are producing step-wise behaviour of the primary pressure at a later phase of transient, while in the measured data pressure level stays below the spray valve opening pressure. This difference may be a result of an inadequate description of heat losses or lack of some plant systems in the calculations. There could for example be a small unmeasured continuous spray at the plant.

In the calculations by VTT, FZR, and NRI, some mixing between the upper plenum and the upper head of the pressure vessel has been modelled by introducing special junctions with small flow rates, between the uppermost part of the pressure vessel and the hot leg. With these junctions the turbulent mixing in axial direction is described, which leads to a cool-down in the upper head. Fig. 7 shows how the calculated upper head temperatures by VTT and FZR follow the hot leg temperatures with some delay. NRI using the same code as FZR (DYN3D /ATHLET) got nearly the same results, so that the curves by NRI and FZR would be practically identical. However, to study the influence of this treatment on the results, NRI carried out a

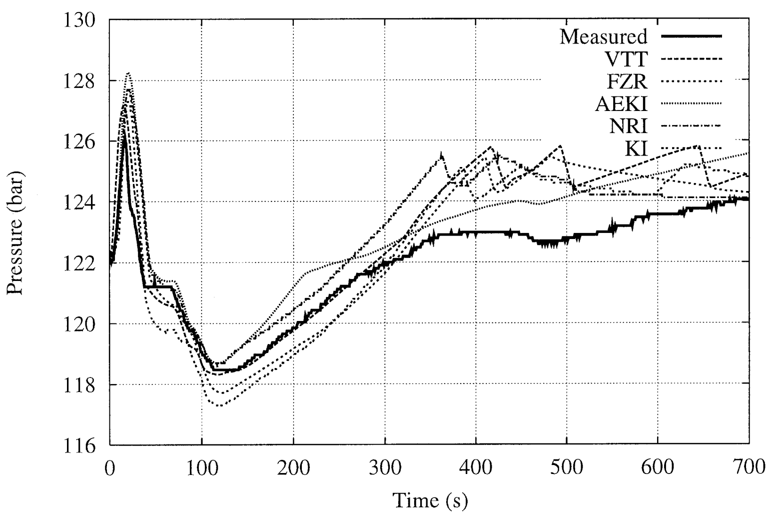

Fig. 6. Primary circuit pressure.

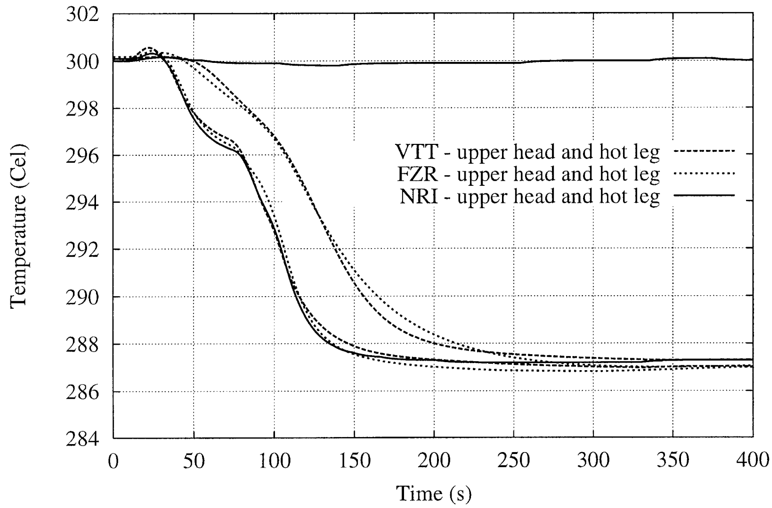

Fig. 7. Calculated upper head and hot leg temperatures.

calculation in which the mixing in the upper head has not been taken into account. This approach resulted in a constant upper head temperature. The larger cool-down in the upper head has some effect on other parameters. The influence can be seen especially in Figs. 8 and 6 as the difference between FZR and NRI results (who calculated the same power level). As a result of the smaller upper head temperature, FZR got a lower primary pressure and pressurizer level than NRI. Despite considering this cool-down by mixing, the pressurizer level is still overestimated.

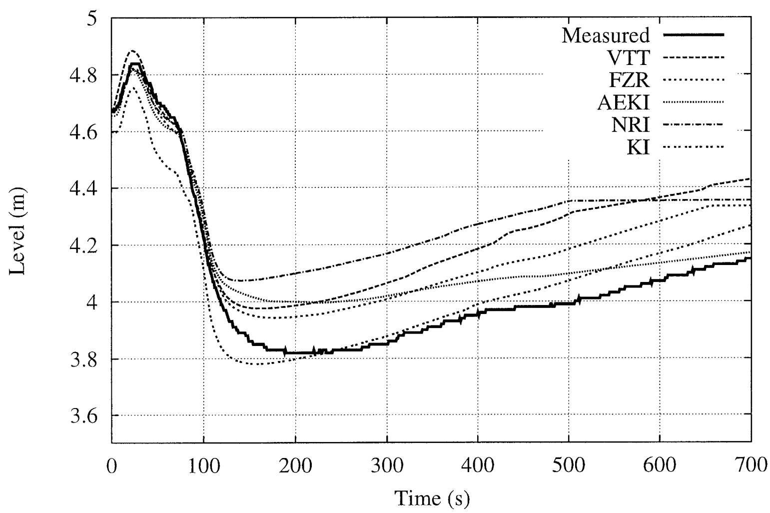

Fig. 8. Measured and calculated pressurizer water level.

5.3. Core

The neutron power starts decreasing due to the control rod group movement. The stepwise measurement of the group position in a VVER-440 reactor (ten steps of 25 cm axially in the core) is shown in Fig. 9. This measurement gives the accurate rod position only at the moment when the value changes. After each step, it is only known that the position of the group is within the 25 cm above the next step. Thus all the tuned insertion histories used in the calculations with different codes are formally consistent with the measurement. As a consequence, this measurement cannot be used for an accurate validation of the control rod reactivity worth. Further uncertainty in predicting the neutron power level is due to the Doppler reactivity feedback effect of the fuel temperature. As in the Balakovo calculations (Mittag et al., 2001), different models and data have been used for the heat transfer from the fuel to the coolant (e.g. gas gap). Hence the fuel temperature results differ considerably between the codes, shown in Table 1. Naturally but unfortunately, there are no direct measurements of the fuel temperature available from the plant.

The neutron powers (Fig. 10) show a rather good agreement with the measured data. The power levels differ significantly between 38 and 70 s, when the control rods have been inserted only in the upper part of the core (about 20% of the core height). However, during this phase, there is a systematic error in the power measurement, because the fast neutron detectors outside the core are located axially at the middle of the core. These detectors do not measure correctly the neutron power decrease occurring in the uppermost part of the core. The magnitude of this error was calculated with HEXTRAN using a new kernel model of the detector. This effect explains half of the 3% discrepancy during this phase in the VTT calculation, the other calculations differed somewhat more. The neutron power measurement is also

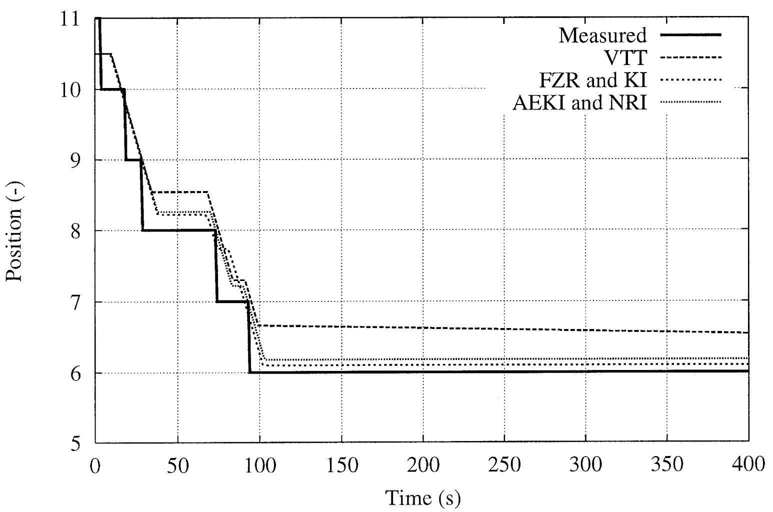

Fig. 9. Position of control rod group 6.

Table 1
Comparison of calculated fuel temperatures at the beginning and at the end of the transient

	VTT	FZR	AEKI	NRI	KI
Core total power (MW),					
beginning	1502.7	1497.3	1503.5	1498.1	1493.0
end	917.6	908.3	898.4	921.0	911.7
Average fuel temperature (°C),					
beginning	585	664	507	745	868
end	469	517	407	573	701
Max fuel temperature (°C),					
beginning	1030	1232	848	1238	1339
end	823	973	681	993	1075

vulnerable to other disturbances, e.g. the changing coolant temperature in the downcomer influences neutron transmission.

The thermocouples measuring the coolant temperatures are encapsulated in tubes which influence the time behaviour of the measured signals. Unfortunately, the time constants of the delayed response of the thermocouples are not well known. However, the importance of taking into account the time constants is illustrated in Fig. 11. The assembly outlet temperature measurement at the core position No. 31 (cf. numbering in Fig. 1) is compared with calculations carried out with and without time delay integration, upper and lower curves in Fig. 11, respectively. The time constant of 30 s gave the best compatibility with the measurements. For the cold and hot leg temperature measurements, with different thermocouples, the time constant of 10 s was most suitable. In Fig. 12, the temperature differences over the core calculated from these values are shown. This temperature difference reflects the total power behaviour. It shows a better agreement with the measurement than the neutron power.

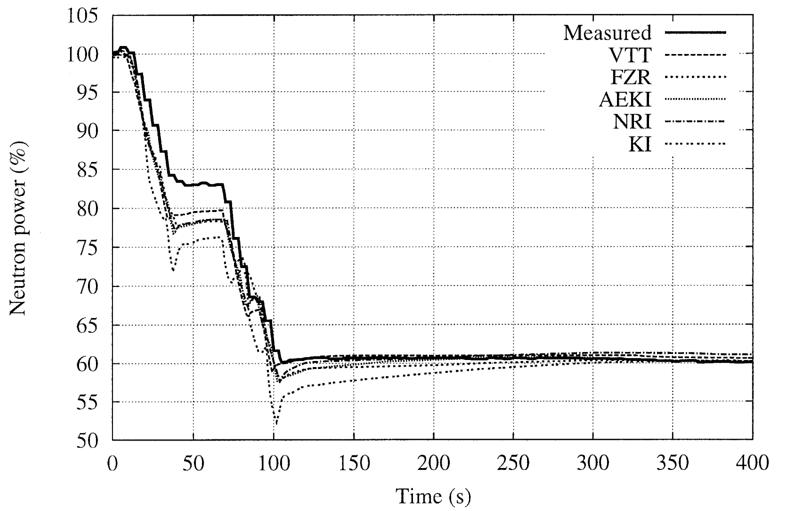

Fig. 10. Neutron power.

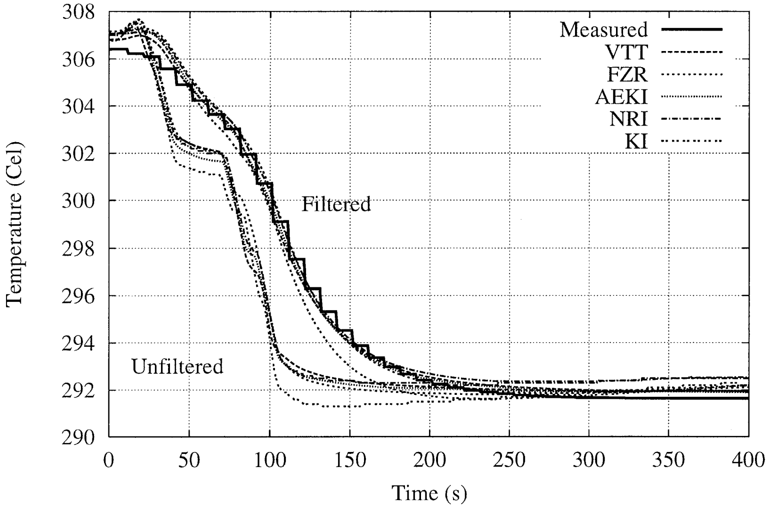

Fig. 11. Fuel assembly outlet temperature at position 31.

The signals of the local rhodium emitter self-powered neutron detectors (SPND) inside the core have not been used earlier in dynamic comparisons because they are quite slow. But as the different half-life components of their delayed responses are known, the detector delay behaviour was simulated in the calculated neutron fluxes, assuming a constant SPND sensitivity during the transient. Hence, the calculated values were normalized at the initial state. Four fuel assemblies of a 60° core symmetry sector are equipped with SPND detector lances, each carrying SPNDs at the distances

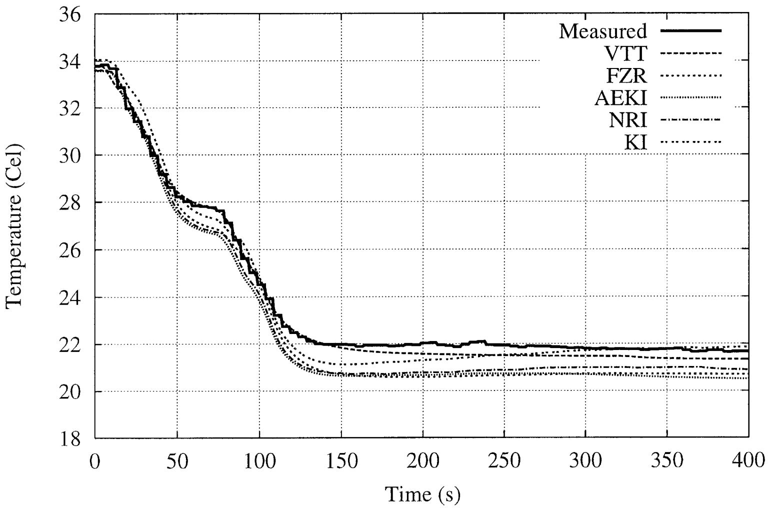

Fig. 12. Average temperature difference between hot and cold legs.

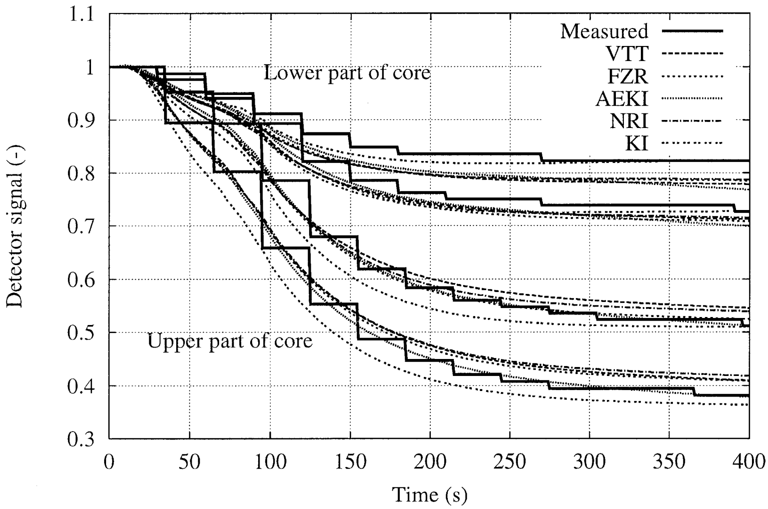

Fig. 13. Relative neutron detector signals in position 53.

of 50, 100, 150 and 200 cm from the active core bottom. In Fig. 13, the time behaviour of the relative signal values from the detectors at these four height positions are shown for the fuel assembly position No. 53 (numbering given in Fig. 1). The different behaviour of the neutron power at different axial levels due to the partly inserted control rods is clearly seen. The agreement with the measurements is satisfying. A similar degree of agreement has been observed for the other three assemblies equipped with SPNDs. In all, the simulation by the codes for the upper core region is better than for the lower part.

6. Conclusions

As well as in part 1 of this work (Mittag et al., 2001), as a result of validation work in the SRR1/95 project, the physical behaviour of the NPPs, especially of the core and the primary circuit is well described by the coupled codes involved. A good agreement between calculated and measured safety-relevant parameters has been achieved for both the VVER-440 and the VVER-1000 reactor transient. In reactor physics calculations the achieved accuracy was somewhat better in the VVER-440 core which comprises smaller and simpler fuel assemblies.

The extent of plant model is important while analysing the real plant data with a rather small amount of measurements. Further, it would also be preferred to have a possibility to model three-dimensionally some parts of the cooling circuits, e.g. the lower and upper plenum. In VVER type plants the modelling of the horizontal steam generators would also largely benefit from 3-D modelling.

Some deviations of the results can be explained by uncertainties in the measurements, e. g. the control rod positions in the VVER-440. It is important to take into account the systematic errors of the measurements, whenever possible. For example, the time delay properties of measuring devices, like thermocouples and SPND, must be considered. If they are unknown, they can be determined by the comparison of measurement and calculations. This approach was justified for the coolant temperatures, because of the good agreement between the calculations by different codes.

During the transient, a main effect was the fission power decrease due to control rod group insertion. Differences in the nuclear data used in the calculations are the cause of different control rod efficiencies leading to differences in power levels. The fuel temperature feedback on reactivity also influences the power. Both effects can hardly be separated. For separately studying the control rod efficiencies, the neutronic codes including the applied nuclear data bases should be additionally be validated against measurements of real plants or research reactors in zero-power conditions. As for the calculated fuel temperature, the present investigations have revealed that it is very sensitive to the modelling of the gas gap between fuel pellets and rod cladding. A dynamic modelling of the gap width is necessary.

Further, it was shown that the coupled neutron kinetic/thermal-hydraulic codes under consideration are capable of simulating typical VVER plant transients. Hence, the work contributes to increase the confidence in the results of the code systems. In future, some more details of the automatic control systems should be included in the thermal-hydraulic system models or plant simulators. The remaining transients, measured in VVER-1000 and VVER-440 plants, that were documented for the SRR1/95 project but not yet calculated by the coupled codes, should be used for further validation work.

Acknowledgements

The Commission of the European Community (CEC) is acknowledged for funding the work in the SRR1/95 project within its Phare programme. The representatives of the beneficiaries of this Phare project: A. Miasnikov (SONS, Czech Republic), S. Stoyanov (CUAE, Bulgaria), I. Lux (HAEC, Hungary), and M. Kristov (NRASR, Slovak Republic) are acknowledged for their interest and useful discussions. The project team members U. Grundmann, U. Rohde, A. Seidel (FZR), E. Kaloinen (VTT), W. Luther, K.-D. Schmidt (GRS), J. Macek (NRI), A. Kereszturi, C. Maraczy (AEKI), M. Lizorkin (KI) and H. Kantee (Fortum Engineering Ltd, Finland) are acknowledged for their work and helpful discussions.

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In: Annals of Nuclear Energy 2001. Vol. 28, No. 1, pp. 857–873.

annals of
NUCLEAR ENERGY

PERGAMON Annals of Nuclear Energy 28 (2001) 857–873
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Validation of coupled neutron kinetic/thermal–hydraulic codes. Part 1: Analysis of a VVER-1000 transient (Balakovo-4)

S. Mittag [a,*], S. Kliem [a], F.P. Weiß [a], R. Kyrki-Rajamäki [b], Hämäläinen [2], S. Langenbuch [c], S. Danilin [d], J. Hadek [e], G. Hegyi [f], A. Kuchin [g], D. Panayotov [h]

[a] *FZR Forschungszentrum Rossendorf, PO Box 510119, D-01314 Dresden, Germany*
[b] *VTT Energy/Nuclear Energy, PO Box 1604, FIN-02044 VTT, Finland*
[c] *GRS, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH Forschungsgelände, D-85748 Garching, Germany*
[d] *KI- Russian Research Center "Kurchatov Institute", Nuclear Reactors Institute, Kurchatov Square 1, 123182 Moscow, Russia*
[e] *NRI Nuclear Research Institute Rez plc, 250 68 Rez, Czech Republic*
[f] *AEKI- KFKI Atomic Energy Research Institute, PO Box 49, H-1525 Budapest, Hungary*
[g] *STCNRS- Scientific & Technical Centre on Nuclear and Radiation Safety, Kharkovskoe shosse 17, 253160 Kiev, Ukraine*
[h] *INRNE- Institute of Nuclear Research and Nuclear Energy, Tzarigradsko shousse 72, Sofia 1784, Bulgaria*

Received 13 July 2000; accepted 15 August 2000

Abstract

Three-dimensional hexagonal reactor dynamic codes have been developed for VVER type reactors and coupled with different thermal–hydraulic system codes. In the EU Phare project SRR1/95 these codes have been validated against real plant transients by the participants from several countries. Data measured during a test in the Balakovo-4 VVER-1000 have been analysed by coupled codes. In the test, one of two working feed water pumps of the steam generators was switched off at nominal power. The steady-state assembly powers measured before and after this transient are reproduced by the codes with a maximum deviation of about 5%. The time behaviour of the most safety-relevant parameters, such as total fission power, coolant temperatures and pressures is well modelled. Thermal–hydraulic feedback effects observed in the measurement are described by the codes in a consistent manner. The analyses have shown, that an accurate treatment of the heat transfer from the fuel rods to the

* Corresponding author.Tel.: +49-351-260-2094;Fax: +49-351-260-2383.
E-mail address: s.mittag@fz.rossendorf.de (S. Mittag).

PII: S0306-4549(00)00095-5

coolant is important. In all, the results have increased the confidence in the coupled code analyses of VVER-1000 transients.

1. Introduction

In recent years, coupled neutron kinetic/thermal–hydraulic codes have been developed, which allow to simulate transients in Russian pressurized water type VVER reactors. These coupled codes can be used for the analysis of the whole reactor system, since they are capable of describing the feedback effects in a consistent manner. The stand-alone neutronic codes, including their nuclear data bases, have already been validated, e.g. by using data from experiments in critical facilities and zero-power reactors, in which there is no thermal–hydraulic feedback on neutron kinetics. The thermal hydraulic system codes have also been separately validated against a lot of experiments in thermal–hydraulic test facilities according to validation matrices. Data for the validation of coupled codes can only be obtained from transients observed in nuclear power stations. The simulation of the plant transients includes the validation of the whole neutronic calculation system, i. e. nuclear data, burn-up calculations, and reactor kinetics. The right modelling of temperature feedback effects by the coupled codes is important.

In the present work (part 1), the data measured in a test carried out in the Balakovo-4 VVER-1000 (Russia) have been analysed by several coupled codes. The results of the code validation against a VVER-440 transient, that was measured in the Finnish NPP Loviisa, will be discussed in part 2.

The coupled-code validation activities for VVERs have been funded by the European Commission in the Phare project SRR1/95.[1]

2. Coupled codes for VVER reactors

In the Institute of Safety Research of Forschungszentrum Rossendorf (FZR), Germany, the code DYN3D (Grundmann et al., 1995) has been developed and coupled to the thermal-hydraulic system code ATHLET, developed by Gesellschaft für Anlagen- und Reaktorsicherheit (GRS), Germany (Teschendorff et al., 1996). VTT Energy, Finland, has coupled its neutron kinetic code HEXTRAN (Kyrki-Rajamäki, 1995) with its thermal–hydraulic system code SMABRE (Miettinen, 1998). Kurchatov Institute (KI), Russia, and Atomic Energy Research Institute (AEKI), Hungary, developed their own neutron kinetic codes BIPR8 (Lizorkin et al., 1992) and KIKO3D (Hegyi et al., 1998), respectively. Both BIPR8 and KIKO3D have been coupled to ATHLET.

[1] The Phare Programme is a European Union initiative which supports the development of democratic nations within a prosperous and stable Europe. Phare does this by providing grant finance to support the process of economic transformation and to strengthen newly created democratic societies. The aim of the SRR1/95 project was to improve the verification of coupled neutron-kinetic/thermal–hydraulic codes in Czech Republic, Bulgaria, Hungary, and Slovak Republic. The facts, opinions and views of this article are the authors' and not necessarily the Commission's.

The system codes ATHLET and SMABRE include basic modules for thermal hydraulics, heat transfer, heat conduction, and one-dimensional neutron kinetics/point kinetics. The thermal–fluid dynamics is described by five-equation models. A six-equation model is additionally available in ATHLET, but has not been used in the present transient analyses.

All the neutronic codes involved are three-dimensional two-group neutron diffusion codes for hexagonal fuel assembly geometry, using fast nodal expansion methods. DYN3D and KIKO3D additionally allow the calculation of rectangular core geometry. Except for BIPR8, the codes include their own fuel rod models and thermal–hydraulic models of the reactor core.

The neutronic codes have been coupled to the thermal–hydraulic system codes in three different ways. In the *external* coupling, the whole reactor core is cut out of the system-code plant model. The core behaviour is completely simulated by the neutron kinetic code which also models the fuel rods and the core thermal-hydraulics. In the *internal* coupling, only the three-dimensional neutron kinetics of the neutronic code is implemented into the thermal–hydraulic system code, which calculates the thermal-hydraulics of the whole system including the core. The third way is a *parallel* coupling. The thermal–hydraulic system code also calculates the whole thermal-hydraulics of the loops and the core but additionally, the neutronic code performs the detailed thermal hydraulics and fuel heat transfer calculation in every fuel assembly of the core, to get the nodal fuel and coolant conditions for the calculation of three-dimensional neutron kinetics and reactivity feedback effects.

DYN3D, BIPR8 and KIKO3D are coupled to ATHLET by the internal coupling method. There is also an external coupling version of DYN3D to ATHLET. The parallel coupling method was applied for HEXTRAN and SMABRE and, additionally, for KIKO3D and ATHLET.

The coupled neutron kinetic/thermal–hydraulic codes listed in Table 1 have been validated against the Balakovo-4 VVER-1000 transient. DYN3D/ATHLET was not only applied by FZR, but also by the Institute of Nuclear Research and Nuclear Energy (INRNE), Bulgaria, and the Scientific and Technical Centre on Nuclear and Radiation Safety (STCNRS), Ukraine.

In all cases, the nuclear data libraries and burn-up codes that were used for the burn-up calculations are consistent with the libraries and neutronic codes applied in the transient analyses.

Table 1
Participating organisations and codes applied

Organisation	Coupled code	Type of coupling	Code for generation of nuclear data
VTT	HEXTRAN/SMABRE	Parallel	CASMO (Ahlin et al., 1977)
KI	BIPR8/ATHLET	Internal	KASSETA (Novikov et al., 1990)
FZR	DYN3D/ATHLET	External	NESSEL (Heinrich, 1981)
INRNE	DYN3D/ATHLET	Internal	NESSEL
STCNRS	DYN3D/ATHLET	External	NESSEL

3. Validation strategy

Transients and accidents to be analyzed by coupled codes are determined by a strong coupling of the neutronics and the fluiddynamics of the primary circuit. Such problems cannot be modelled properly by a separate application of the stand-alone codes, because the definition of time-dependent boundary conditions would introduce unacceptable uncertainties.

Taking into account previous validation activities in the neutronic and thermal–hydraulic field together with the specific features of coupled codes, emphasis should be put on the following aspects in the validation process:

- Neutronics effects on plant transient behaviour,
- fluiddynamics effects on reactor core behaviour,
- control system effects on reactor core behaviour

The modelling of the control system needs specific consideration and illustrates some peculiar aspects of code validation. In a realistic plant model the complete control system should be included. The input preparation for such models needs a great number of parameters to describe all controller elements. Practically, the detailed information needed for such a complete model can only be provided by the plant designer. In transients, that are interesting for code validation, very often parts of the control system fail or operate in abnormal conditions. Therefore, the functions of the control system must be partially replaced by time-dependent boundary conditions. Typical examples are: The control of the feed water mass-flow rate in the steam generator as a function of the actual water level is replaced by a measured time function of the mass-flow rate. The movement of the control rods activated by the automatic power control is replaced by the time-dependent positions of the control rods. Such replacements of functions of the control system are acceptable, but they should be carefully described.

In the present work, priority has been given to the simulation of the physical behaviour of the reactor, especially of the core and the primary circuit. Measured safety-relevant parameters, such as power, coolant temperatures, pressurizer water levels and pressures, have to be compared with the calculations. The simulation of the secondary circuit and of the plant controllers was of minor interest.

4. The Balakovo-4 VVER-1000 test

In the test, one of the two working steam generator feed water pumps was switched off at nominal power. Two seconds after the pump switch-off, the power control system responded by inserting the control rod group K1 from core top to bottom within four seconds. Group K10, that had been at the axial position of 275 cm, started moving in at the rate of 2 cm/s. As a result of dropping group K1, the neutron power decreased to about 63% of the nominal value within 10 s. The slow insertion of group K10 down to a position of 140 cm resulted in a power decrease to about 45%. The reactor power was stabilised at this level by the automatic power controller. The differences between the temperatures of the hot legs and the corresponding cold legs of the four primary circuit loops decreased proportionally to the thermal power reduction, as all four pumps continued working. In the secondary circuit, where the transient had been initiated, the flow rate through the second feed water pump that was still in operation, was increased by some 50% within 16 seconds after switching off the first pump, in order to partly compensate the deficient feed water flow. In the following, the flow rate of the second pump was reduced again to match the decreasing thermal power of the primary circuit. During the whole transient, the water level in the steam generators was always kept well above the heater tubes.

4.1. Burn-up calculations

To obtain the reactor state before the transient, depletion calculations have been carried out by all participants over the first 152 full-power days of the first fuel cycle, using the given operation history. Fig. 1 shows the calculated assembly burn-up values for a 60° symmetry sector of the reactor core. The results by VTT were used

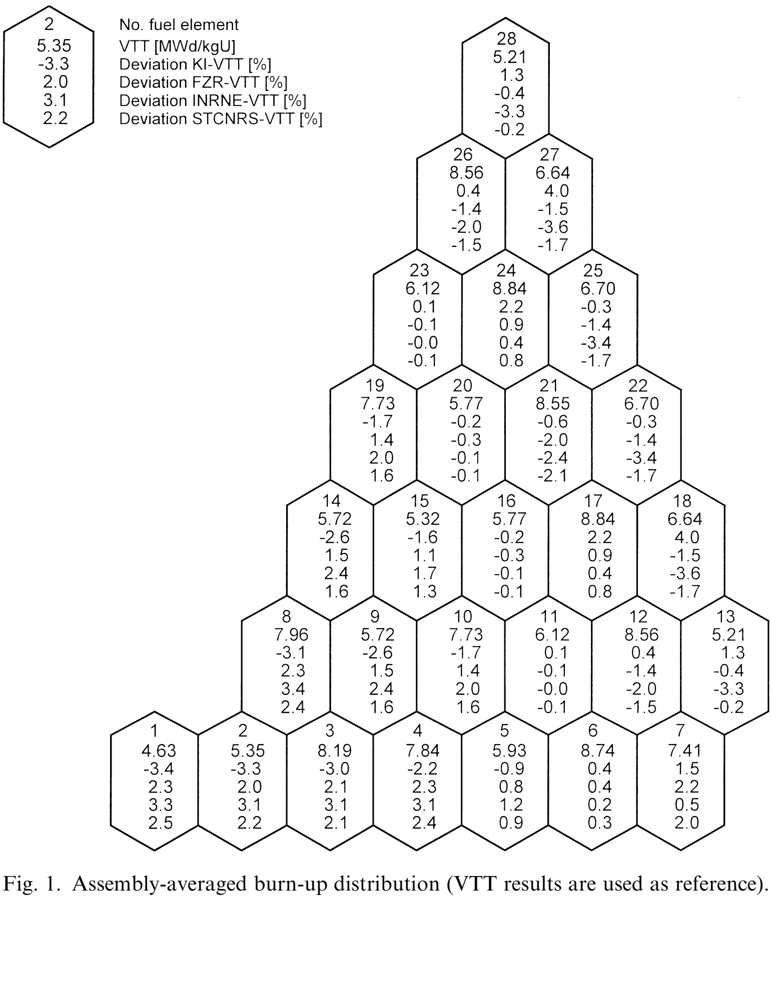

Fig. 1. Assembly-averaged burn-up distribution (VTT results are used as reference).

as an arbitrary reference for the comparison. The maximum deviations of about 4% occur in assemblies near to the boundary of core. They are mainly caused by different boundary conditions (albedos). A higher neutron reflection at the radial core edge in the KI calculations leads to a higher burn-up there.

The critical boric acid concentrations calculated for the nominal-power state after 152 full-power days are compared with the measured value in Table 2.

The measurement is best met by the VTT calculation. The DYN3D calculations (FZR, INRNE, STCNRS) underestimate the measured critical boric acid concentration. Small deviations between the three calculations with the DYN3D code and the same nuclear data library may be caused by slightly different assumptions in the calculation conditions (e.g. core coolant bypass flow rate, albedo values at the core boundary, axial mesh). The KI° calculation overestimates the measured value, which is partly due to the higher neutron reflection. Additional calculations and comparisons of *k*-infinity values for the fuel assemblies indicate that the deviations in Table 2 are mainly caused by differences in the nuclear cross section data. Nevertheless, all the calculated critical boric acid concentrations are within the accuracy of measurement carried out in the Balakovo NPP, which amounts to ±0.5 g/kg.

Table 2
Comparison of critical boric acid concentrations

	Boric acid con-centration (g/kg)	Deviation (g/kg)
Measurement	3.00	
KI	3.27	+0.27
VTT	3.04	+0.04
FZR	2.74	−0.26
INRNE	2.78	−0.22
STCNRS	2.78	−0.22

4.2. Steady state of the core before and after the transient

As a first step of validation, comparisons are made between measurements and calculations in steady states. In Table 3, the main data are compiled characterising the stationary core states before and after the transient. The stationary xenon and

Table 3
Main parameters for the steady states before and after the transient

	Before	After
Reactor power (MW)	2949	1320
Axial position of control rod group K1 (cm)	355 (totally moved out)	0 (totally inserted)
Axial position of control rod group K10 (cm)	274	140
Coolant inlet temperature (°C)	287.2	281.5
Coolant flow rate (kg/s)	18 000	18 300

samarium distributions calculated for the state before the transient were also used in the calculations for the state after the transient, assuming that these distributions had changed very little during the transient lasting 10 min.

Three-dimensional core power distributions were provided by the utility for the steady states before and after the transient. The distributions were derived from self-powered neutron detector (SPND) in-core measurements. Fig. 2 shows the comparison of the measured assembly-wise relative power density distribution with the values that were calculated by the different codes. The fuel assemblies marked by thick lines are equipped with control rod clusters. For this reason, they do not carry in-core detectors. The locations of the control rod groups K1 and K10 that were moved during the transient (cf. Table 3) are provided in the figure, the other groups remained always totally out of the core. The maximum deviation between measured and calculated powers is 4.6%, in an assembly at the core boundary. Fig. 3 depicts the assembly powers for the steady state after the transient. As a result of control rod insertion, the relative powers measured in fuel assemblies neighbouring to the

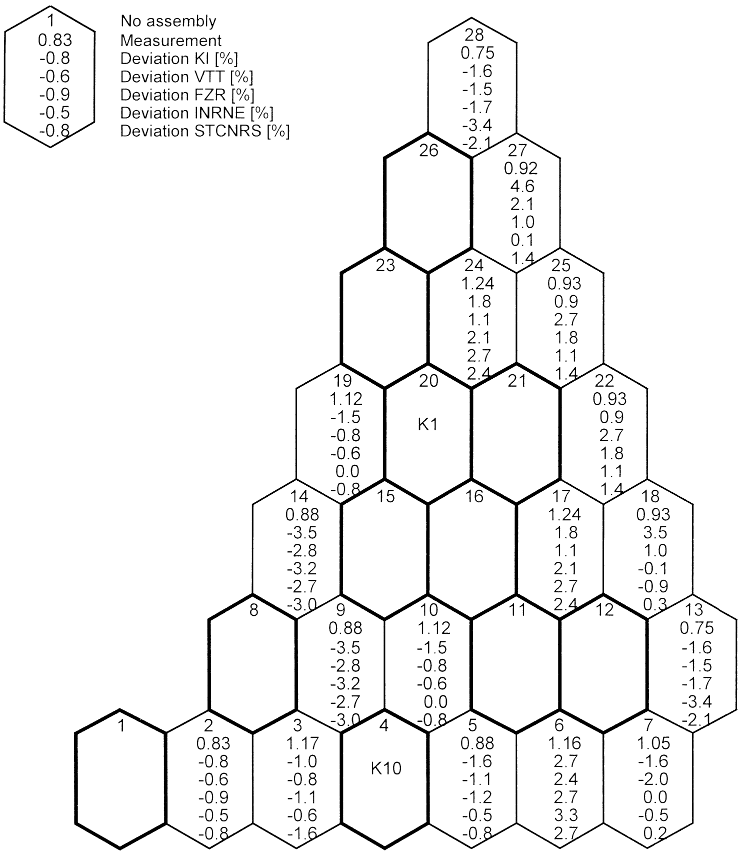

Fig. 2. Assembly-averaged power distribution before the transient, K1, K10: Control rod groups.

groups K1 and K10 are below the corresponding values in Fig. 2. This influence is well described by the codes. As an example for the axial power distributions, the comparison in the hottest fuel assembly (no. 17) prior to the transient is presented in Fig. 4. In all, there is a good agreement. However, there is an over-estimation of the measurement by all calculations near the upper boundary of core.

4.3. *Transient analysis*

4.3.1. *Modelling assumptions*

The measured time-dependent control rod group positions were applied in the calculations. Fig. 5 shows the movement of group K10.

In the ATHLET calculations, a simplified description of the secondary side of the plant was used: The measured pressure in the main steam header (MSH) was applied as a boundary condition. The regulation of the calculated steam mass flow is modelled by a controller, which guarantees, that the MSH pressure in the calculation

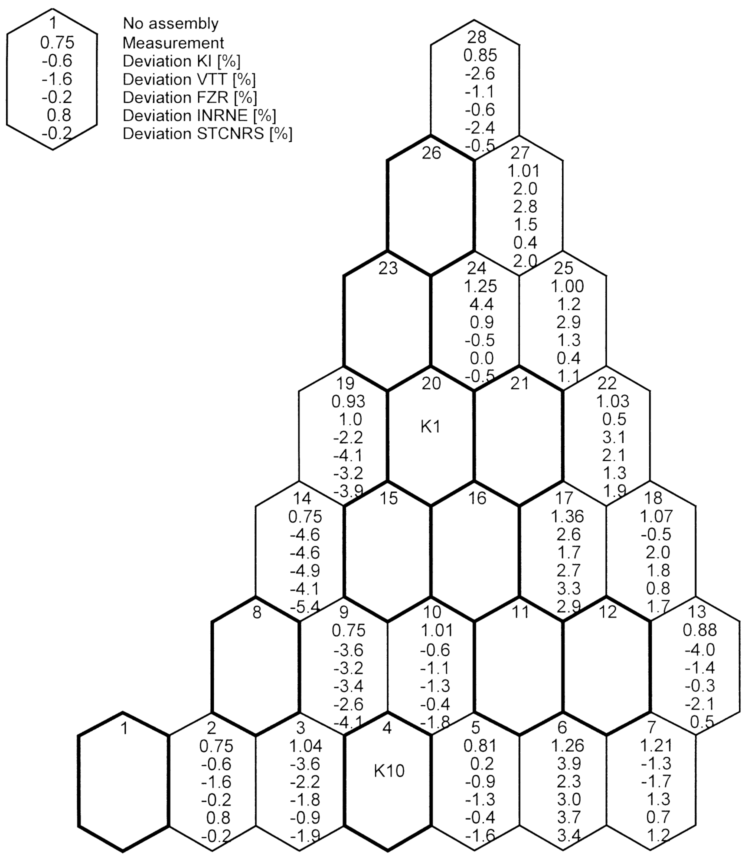

Fig. 3. Assembly-averaged power distribution after the transient, K1, K10: control rod groups.

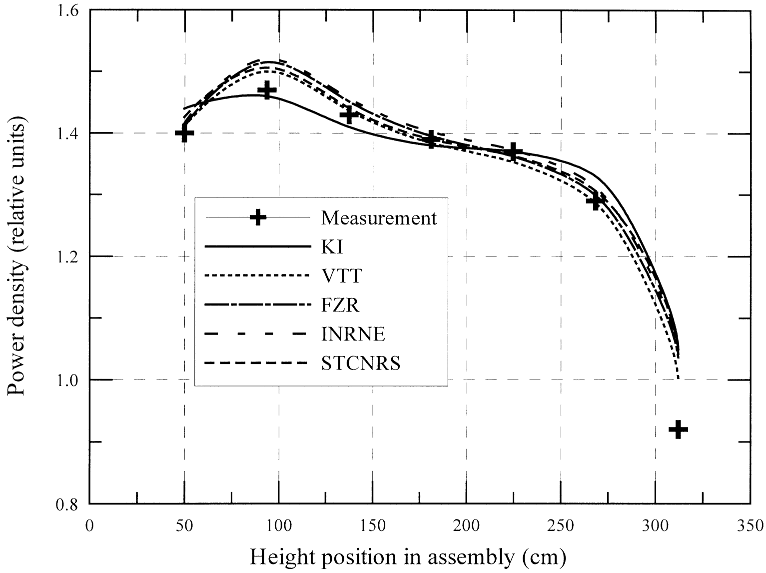

Fig. 4. Axial power distribution of the hottest assembly (no. 17).

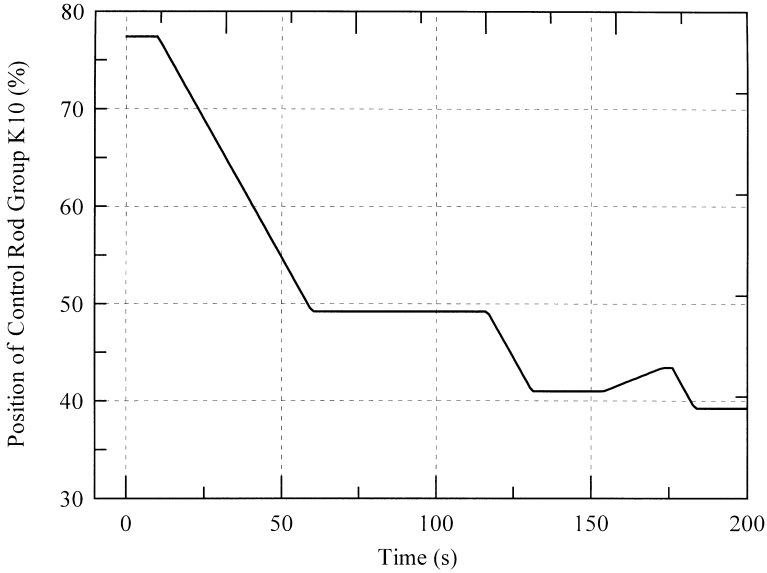

Fig. 5. Movement of control rod group K10.

follows the data of the measurement. The measured feed water mass flow rate was also used as a boundary condition.

In the SMABRE input deck, the secondary circuit was modelled in more detail. The feed water collector as well as the feed water pumps are included in the input deck. The control of the feed water valves is based on the collapsed water level.

In the HEXTRAN/SMABRE calculation, the first group of heaters in the pressurizer is assumed to be not operating. 3% of the primary circuit flow are assumed to pass the upper head, thus cooling the head during the transient. Such an additional circulation is not accounted for in the ATHLET input decks.

4.3.2. Transient results

The transient was initiated by switching off one feed water pump. The following closure of the turbine control valve causes a pressure increase on the secondary side. Fig. 6 depicts the main steam header pressure which is used as boundary condition in the ATHLET calculations. In the SMABRE calculation (VTT), the first pressure peak appears a little earlier due to the different modelling. But this has no influence on the other, especially primary circuit parameters. Later on, the pressure is stabilised by the controller at about 6.0 MPa. The oscillations are smaller than in the ATHLET calculations and in the measurement.

The nuclear power (Fig. 7) is the most interesting parameter, with respect to the interaction between neutron kinetics and thermal hydraulics. At the beginning of the transient, the power decreases very fast as a result of the activation of the rapid unit unloading system (dropping control rod group K1 within 4 s). In the following, the power is mainly determined by inserting control rod group K10.

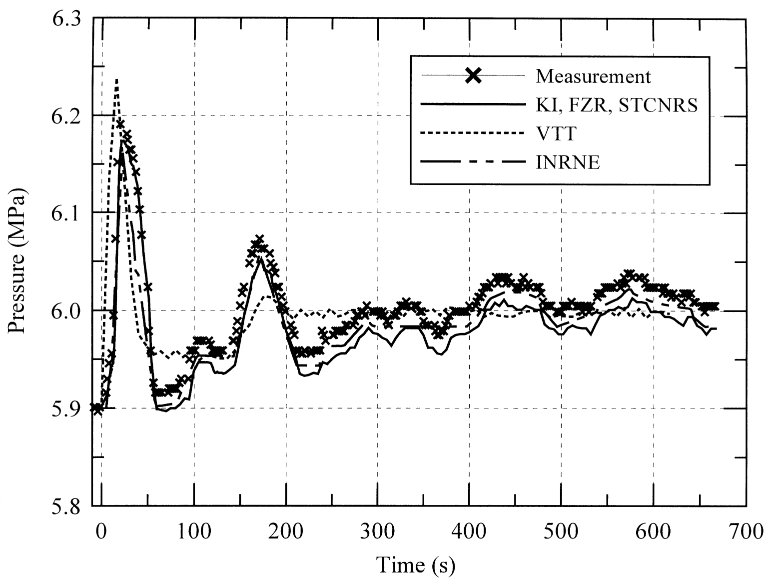

Fig. 6. Main steam header pressure.

At $t = 100$ s, the measured power amounts to 1480W, the calculated values are between 1470 and 1550 MW. After further insertion of K10 the power is finally stabilised at 1320 MW. All calculations overestimate the power at the end of the transient. However, the deviations do not much exceed the neutron power measurement accuracy of 2%.

In Fig. 8, the power behaviour at the transient start is shown in more detail. A power minimum is reached, when K1 arrives at the core bottom. Four calculations (FZR, VTT, INRNE, STCNRS) yield nearly the same minimum value of about 1550 MW, only KI calculated a higher value (1700 MW), which seems to be in best agreement with the measured power minimum. However, the power is measured with a time resolution of 2 s, which does not allow to fix the measured power minimum in this time interval as exactly as in the calculations.

The course of the power observed up to 20 s, is significantly influenced by temperature feedback. The fast decreasing power causes a delayed fuel temperature reduction, which continues after the end of the K1 drop ($t = 6$ s). The Doppler feedback leads to a reactivity growth and a power increase. The control rod group K10 starts moving in at $t = 10$ s. Up to $t = 20$ s, however, the reactivity decrease due to the slow K10 insertion is practically compensated by the reactivity increase caused by the dropping fuel and coolant temperature. Hence, no power reduction by the K10 movement can be observed in this time interval.

The power rise measured immediately after the end of the insertion of control rod group K1 is about 200 MW. The calculated values are between 190 and 300 MW. The deviations between the calculations are caused by different temperature dependencies of the nuclear cross sections and different assumptions on the thermal properties of the fuel rods.

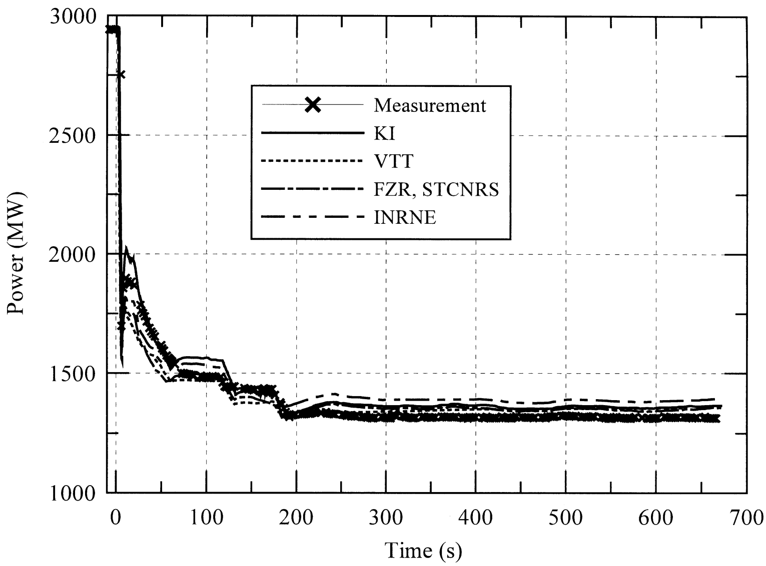

Fig. 7. Nuclear power.

Detailed investigations have shown that the change of the heat transfer coefficient in the gas gap during the transient is relevant. With dropping fuel temperatures, the gap width increases, which leads to a decreasing heat transfer coefficient. This reduces the fuel temperature decrease. Comparing the results of DYN3D/ATHLET calculations with constant (internal coupling) and changing (external coupling) heat transfer coefficient in the gap, performed by FZR, has shown that the difference caused by this effect in the power level of the final state is about 150 MW (Fig. 9).

The pressure in the primary circuit is also an important parameter for the safe reactor operation. Fig. 10 depicts the upper plenum pressure. All calculations provide a similar behaviour. The lowest measured pressure value, reached during the transient is 14.75 MPa. The calculated minimum values are between 14.85 MPa (FZR) and 14.95 MPa (INRNE). This is mostly due to the differences in the power at this time. Later on, the pressure rise is faster in the calculations than in the measurement. The oscillations in the ATHLET calculations, beginning after $t = 220$ s are caused by the oscillations in the secondary circuit pressure. At about 400 s, the switching points of the pressurizer heater groups are reached in the calculations and the pressure remains constant up to the end of the transient. Between $t = 80$ s and $t = 220$ s, the difference between the calculated and the measured curves increases up to 0.4 MPa. At the end of the transient, the deviation is reduced to about 0.1 MPa.

A possible explanation for the discrepancy may be found in the pressurizer heater operation model. Fig. 11 shows the results of additional calculations, where the pressurizer heater groups 3 and 4 were switched off at $t = 80$ s, so that the pressure set points are not reached. The pressure behaviour calculated by all participants up to $t = 300$s is in better agreement with the measured curve than in the basic

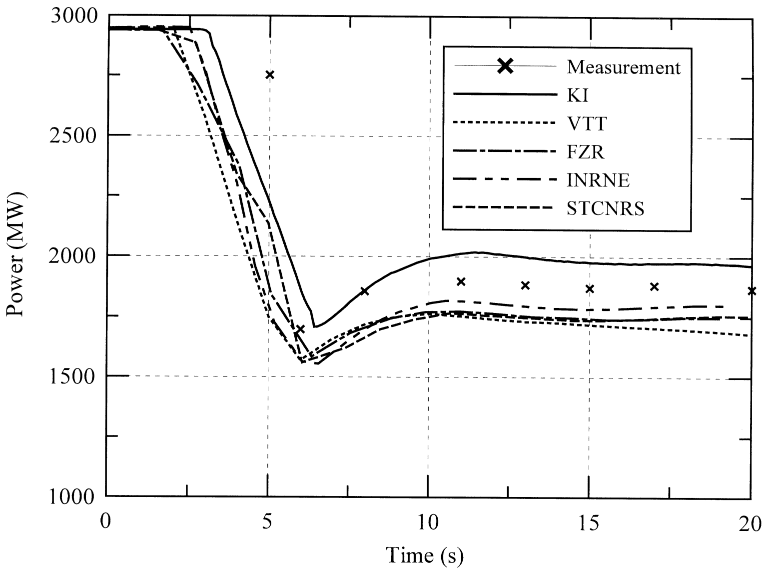

Fig. 8. Nuclear power ($t = 0...20$ s).

calculation (Fig. 10). Later on, the calculated pressure curves show a smaller increase than the measured values. The comparison of Figs. 10 and 11 suggests, that the heater groups 3 and 4 were not continuously in operation. It may be presumed, that there is a similar control signal as known from the Loviisa VVER-440

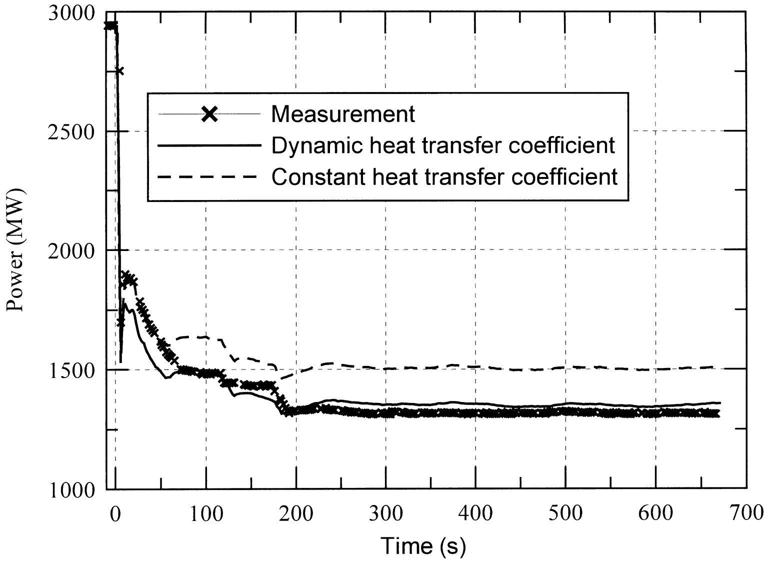

Fig. 9. Nuclear power by DYN3D/ATHLET applying different fuel rod models.

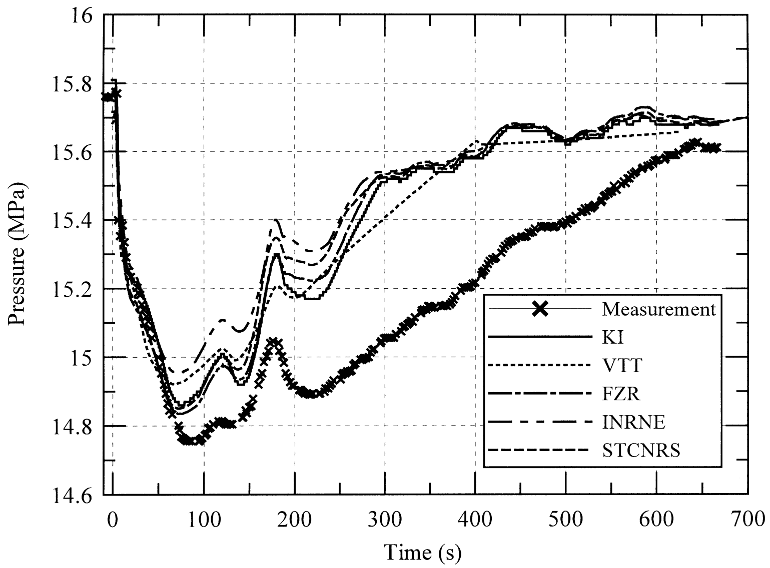

Fig. 10. Upper plenum pressure.

(Finland), which prevents the operation of the highest heater groups during too fast heating-up of the primary circuit. Unfortunately, the real heater operation has not been recorded in the NPP.

The measured collapsed level decreases in the pressurizer during the transient by more than 2 m. Within the first 80 s, all calculations show a faster decrease of the level in comparison to the measured value (Fig. 12). After some 200 s, the level is stabilised at a new stationary value. The KI and the FZR values are practically in full agreement with the measured one. The other calculations differ between 10 and 20 cm. This deviation is acceptable, having in mind that the pressurizer level is kept at its set point with an accuracy of ±15 cm by the coolant volume controller.

For the comparison between measurement and calculation, a time filter for all calculated temperature values was considered. This filter models the delayed response of the thermocouple temperature measurement. The low-pass filter time constant is 16 s. From the beginning up to $t=220$ s, all calculated hot leg temperatures are in good agreement with the measured curve (Fig. 13). From $t=200$s on, the measured hot leg temperatures are below the calculated values. At the same time, the measured power is also overestimated by the calculations (Fig. 7). This is the reason for the deviations between calculated and measured hot leg temperatures, since the behaviour of the measured cold leg temperature during the transient is well modelled in all calculations (see Fig. 14).

The differences in both hot and cold leg temperatures are within the measurement accuracy of 1–2 K.

The peak in the cold leg temperature at about 200 s reflects the influence of the secondary circuit pressure, which has also a peak at this time (Fig. 6). This pressure

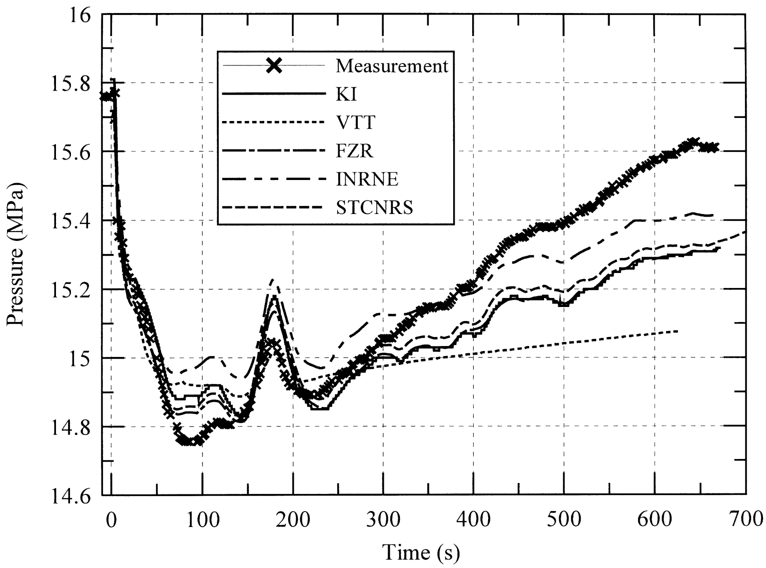

Fig. 11. Upper plenum pressure (modification of pressurizer heater operation).

is controlled for the most part by the turbine valves. A rising pressure at the secondary side of the steam generator causes an increasing saturation temperature, which leads to a diminished heat transfer in the steam generator. The time shift of the measured cold leg temperature peak against the pressure peaks in the sec-

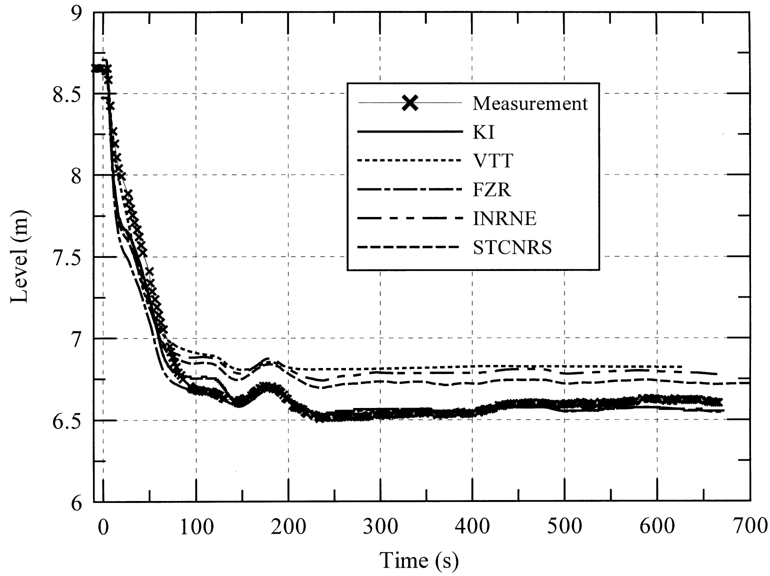

Fig. 12. Pressurizer level.

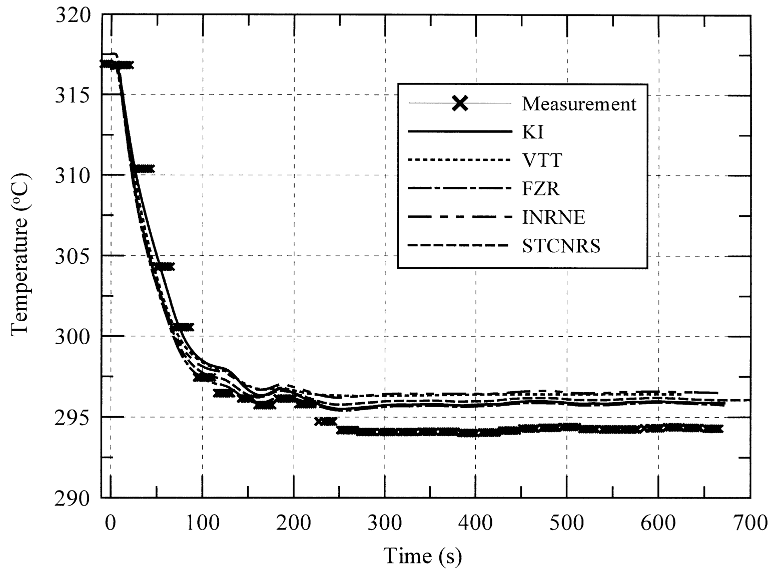

Fig. 13. Hot leg temperature.

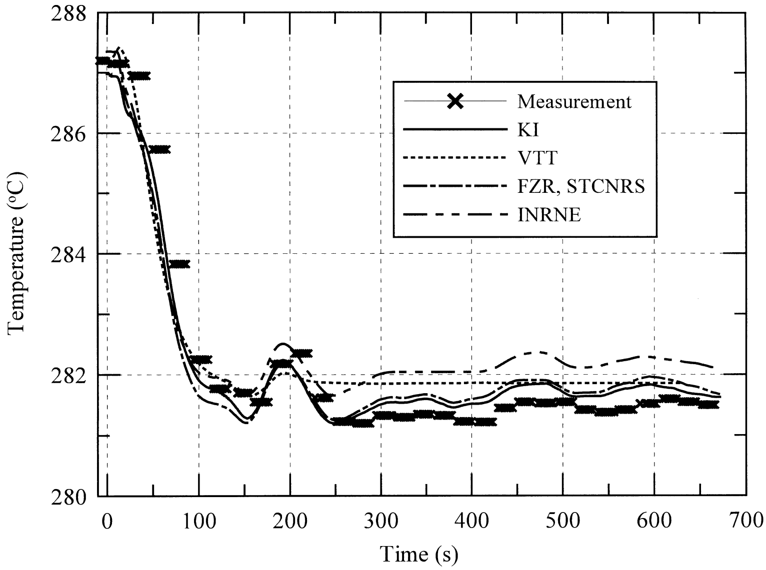

Fig. 14. Cold leg temperature.

ondary and primary circuit (see also Fig. 10) is due to the delayed response of the thermocouples.

5. Conclusions

Generally, the physical behaviour of the Balakovo-4 VVER-1000, especially of the core and the primary circuit is well described by the coupled codes involved. A good agreement between calculated and measured safety-relevant parameters has been achieved. The interaction between neutron kinetics (neutron power) and thermal hydraulics that can be observed in the measurement is modelled in a consistent manner by all coupled codes involved.

The deviation between calculated and measured primary pressure can be explained by uncertainties in the measurement, i.e. the lack of information on the real pressurizer heater operation.

The calculated fuel temperature has turned out to be sensitive to the modelling of the gas gap between fuel pellets and rod cladding. Hence, a dynamic treatment of the gap width is necessary.

For future coupled code validation, an uncertainty analyses would be very useful, studying the quantitative influence of several sources of uncertainty, e.g. in the parameters used in the models and the reactor operating conditions.

Acknowledgements

The Commission of the European Community (CEC) is acknowledged for funding the work in the SRR1/95 project within its Phare programme. The representatives of the Beneficiaries of this Phare project: A. Miasnikov (SONS, Czech Republic), S. Stoyanov (CUAE, Bulgaria), I. Lux (HAEC, Hungary), and M. Kristov (NRASR, Slovak Republic) are acknowledged for their interest and useful discussions. The project team members U. Grundman, U. Rohde, A. Seidel (FZR), E. Kaloinen (VTT), W. Luther, K.-D. Schmidt (GRS), J. Macek (NRI), A. Kereszturi, C. Maraczy (AEKI), M. Lizorkin (KI), V. Khalimonchuk (STCNRS), B. Ilieva (INRNE) and H. Kantee (Fortum Engineering Ltd, Finland) are also acknowledged for their work and helpful discussions.

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In: Proceedings of Nuclear Safety Research Conference (NSRC).
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FRAPTRAN Fuel Rod Code and its Coupled Transient Analysis with the GENFLO Thermal-Hydraulic Code

Keijo Valtonen[1], Anitta Hämäläinen[2], Mitchel E .Cunningham[3]

[1]Radiation and Nuclear Safety Authority, STUK, P.O. Box 14, FIN-00881 Helsinki, Finland
[2] VTT ENERGY, P.O. Box 1604, FIN-02044 VTT, Finland
[3]Pacific Northwest National Laboratory, P.O. Box 999, Richland, WA

ABSTRACT

The FRAPTRAN computer code has been developed for the U.S. Nuclear Regulatory Commission (NRC) to calculate fuel behavior during power and/or cooling transients at burnup levels up to 65 MWd/kgM. FRAPTRAN has now been assessed and peer reviewed. STUK/VTT have coupled GENFLO to FRAPTRAN for calculations with improved coolant boundary conditions and prepared example calculations to show the effect of improving the coolant boundary conditions.

INTRODUCTION

Plans to increase the burnup of nuclear fuel, to utilize new fuel designs, and in some countries to include additional transients such as the anticipated transient without scram (ATWS) in safety evaluations, require that new or updated models be used in safety analyses. Addressing these issues requires improving the fuel models in reactor dynamics codes or incorporating more advanced thermal-hydraulic models in fuel behaviour codes. An example of the latter approach involves coupling the Finnish thermal-hydraulic model GENFLO (GENeral FLOw) with the new US Nuclear Regulatory Commission (NRC) FRAPTRAN code. Provided in this paper are a summary of the recent code assessment and peer review of FRAPTRAN, a description of GENFLO and its coupling with FRAPTRAN, and the results of two example calculations using FRAPTRAN-GENFLO.

FRAPTRAN ASSESSMENT AND PEER REVIEW

FRAPTRAN (Fuel Rod Analysis Program TRANsient) is being developed and maintained for the NRC to calculate fuel behavior during power and cooling transients at burnup levels up to at least 65 GWd/MTU (Cunningham et al. 2001a). This FORTRAN-language code will be applied for the evaluation of fuel behavior during transients such as reactivity accidents, loss-of-coolant-accidents, and boiling-water reactor power oscillations without scram. FRAPTRAN uses a finite difference heat conduction model for the transient thermal solution, the FRACAS-I mechanical model, and the

MATPRO material properties package. To account for the effects of high burnup, FRAPTRAN uses a UO_2 thermal conductivity model that incorporates the degradation effects of burnup and a revised model for Zircaloy mechanical properties that accounts for the effect of oxidation and hydrides in addition to irradiation damage. Burnup dependent fuel rod initial conditions can be obtained from the companion FRAPCON-3 (Berna et al. 1997) steady-state fuel rod performance code.

FRAPTRAN has been assessed (Cunningham et al. 2001b) using a data base that emphasized experiments investigating the effects of burnup on fuel rod behavior during reactivity-initiated accidents (RIAs) and loss-of-coolant-accidents (LOCAs). FRAPTRAN generally performed well in the comparisons to data. Principal conclusions for the code-data assessment include:

- Comparison of code predictions with data have provided assurance that the basic models are working satisfactorily; i.e., temperature, gap conductance, gas pressure, and thermal expansion.

- Comparisons of predicted and measured fuel centerline temperature during scrams show that the code consistently calculates faster temperature decreases than were measured. This is likely due to FRAPTRAN calculating lower thermal resistances in the fuel or fuel-cladding gap than are operating in the rods.

- Rod internal gas pressure is correctly calculated when other paramters that determine gas pressure, such as available volume and corresponding temperatures, are correctly input and calculated. In addition, when gas pressure is correctly calculated for the LOCA cases, then reasonable agreement between predicted and measured time to failure is obtained; this is illustrated in Figure 1 for the MT-4 experiment (Wilson et a. 1983). The initial decreases in pressure are due to cladding ballooning and rod failure is indicated by pressure decreasing to system pressure (i.e., 0.28 MPa).

- FRAPTRAN provides reasonable predictions of cladding axial elongation for fast transients but, as expected because the code does not include a fuel creep model, does not follow the fuel and cladding relaxation when steady-state power conditions are achieved; this is illustrated in Figure 2 where measured and predicted cladding elongation are compared for the IFA-508 test in the Halden Boiling Water Reactor (Uchida and Ichikawa 1980). For this experiment, power was held constant after each power increase, and cladding elongation could be seen to decrease. FRAPTRAN compared well to the rate of cladding elongation for each power increase, but does not calculate the relaxation during the power hold.

- FRAPTRAN consistently underpredicts permanent cladding hoop strain for the RIA tests recently conducted in the NSRR. This is indicative of fuel-cladding mechanical interaction occurring in these tests that is not modeled by the code. This underprediction of permanent hoop strain is illustrated in Figure 3 by comparing predicted hoop strain to measured hoop strain for tests conducted in the NSRR (Fuketa, Nakamura, and Ishigima 1998) and the CABRI test facilities (Papin and Schmitz 1998).

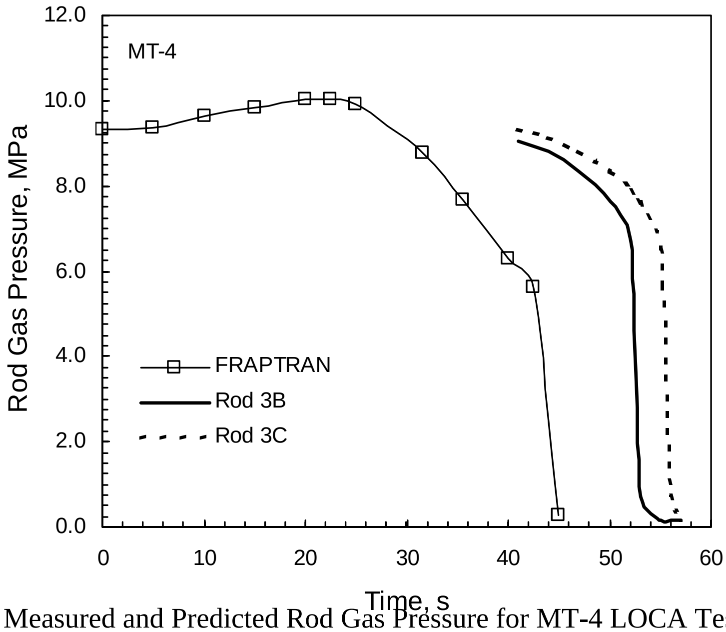

FIGURE 1. Measured and Predicted Rod Gas Pressure for MT-4 LOCA Test

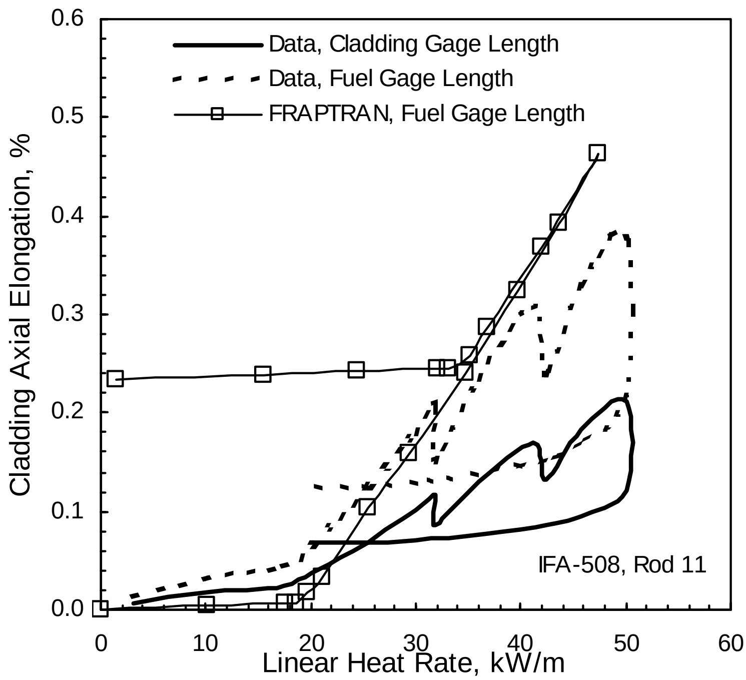

FIGURE 2. Comparison of Predicted and Measured Cladding Elongation for IFA-508, Rod 11

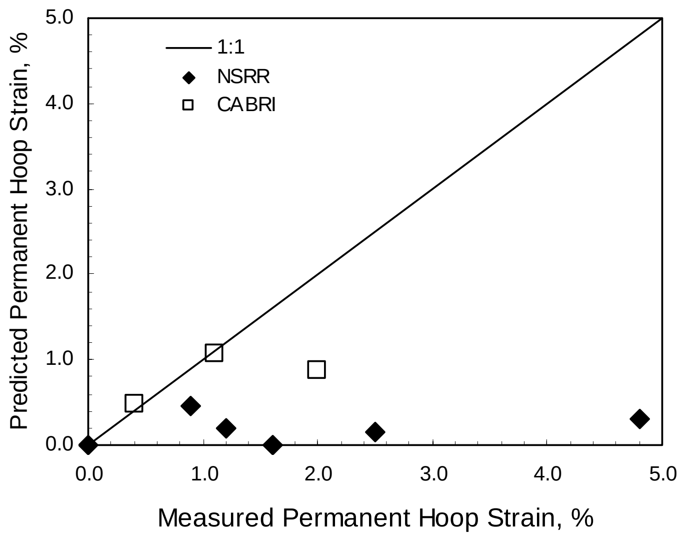

FIGURE 3. Comparison of Measured and Predicted Permanent Cladding Hoop Strains for RIA Experiments Conducted in the NSRR and CABRI Facilities

In addition to the code assessment, an independent peer review of FRAPTRAN was conducted using peer reviewers representing a broad background of code developers, code users, and experimentalists. Each reviewer had greater than 20 years of experience. After reviewing the draft description and assessment documents, and meeting and discussing their comments with FRAPTRAN's developers, many useful recommendations were made and implemented. Principal conclusions from the peer review process include: the FRAPTRAN fuel and cladding models are reasonable, FRAPTRAN is able to predict the trends in the transient experimental data, and the assessment data base adequately covered the intended applications for the code. Recognizing that development work will continue on FRAPTRAN, the reviewers recommended that priority be given to adding models for transient fission gas release and transient fuel swelling/creep.

FRAPTRAN is being released though the FRAPCON-3 and FRAPTRAN users group managed by Pacific Northwest National Laboratory for the NRC. Members of the users group will be supplied the documentation, the code, and a selected set of sample problems. Information about the users group can be found on the FRAPCON3 website (www.pnl.gov/frapcon3).

THE GENFLO MODEL AND COUPLING WITH FRAPTRAN

The thermal-hydraulic models provided in FRAPTRAN are applicable for homogeneous, slowly changing thermal-hydraulic conditions and for many transients it is necessary to use a thermal-hydraulic code to calculate coolant boundary conditions. Especially during the ATWS in BWR plants, the hot channel and the whole core may experience rapid transitions between the wetted and dry states. Because of this, a dynamic exchange of detailed local data between the fuel performance and thermal-hydraulic models is needed.

The thermal-hydraulic model GENFLO (GENeral FLOw) has been developed by VTT Energy (VTT) and under the sponsorship of the Finnish Radiation and Nuclear Safety Authority (STUK) the code is coupled with the FRAPTRAN code. The thermal-hydraulic solution principles of GENFLO are based on the models developed for the SMABRE model (Miettinen and Hämäläinen 2000). GENFLO is a fast running, five-equation model, where the wetted wall heat transfer, dryout, post-dryout heat transfer and quenching models are included. The flow modes covered by GENFLO are depicted in Figure 4. The geometry described by GENFLO comprises one or several parallel fluid flow channels and an optional fuel structure. The lower and upper plena are always included but the core bypass and the downcomer may be zeroed, as is done in subchannel applications.

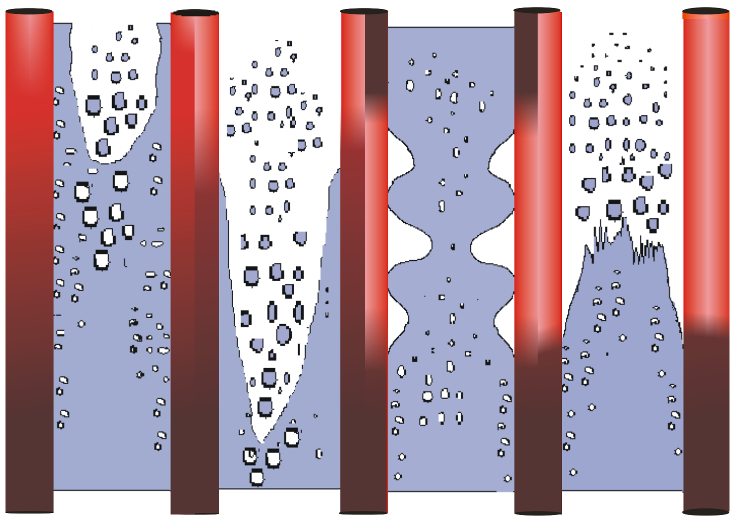

FIGURE 4. The channel flow modes in GENFLO

GENFLO solves the coolant mass, momentum and energy conservation equations, including the calculation of the axial distributions of the fluid temperature and the void fraction. As a result, the fluid temperatures and heat transfer coefficients for each axial level at each time step are supplied for FRAPTRAN. The temperatures in the fuel rod, and the deformation of fuel pellets and cladding, including possible ballooning, are calculated in FRAPTRAN using burnup dependent models. At this stage GENFLO and FRAPTRAN use their own models, i.e., for fuel and cladding temperatures including cladding oxidation and hydrogen generation, though FRAPTRAN supplies the local gas gap heat transfer coefficient for GENFLO. The axial power profile used by both FRAPTRAN and GENFLO comes from the system code. In the future, models now performing in parallel will be unified.

In the coupled code, FRAPTRAN is the master code calling GENFLO which provides the thermal-hydraulic conditions for the whole channel. This calculation is performed only once for each time step, even if a number of iterations would be done in FRAPTRAN during the time step. In the beginning, there is a need from GENFLO for a steady state calculation before any coupled code calculation. In the coupled code calculation, the FRAPTRAN code dictates the length of the time step. A typical time step is short, 0.01 to 0.05 seconds. Even with such short steps, the actual FRAPTRAN-GENFLO calculation is not time-consuming because of the non-iterative feature and effective numerical methods for the fast running thermal-hydraulics module.

The system behaviour and boundary conditions needed for detailed core simulation and studying the fuel rod behaviour with FRAPTRAN-GENFLO may be calculated with various system codes such as RELAP5 or others. The results of the analyses of the three-dimensional BWR or PWR reactor dynamics codes TRAB-3D and the simulator APROS have been used by STUK/VTT. The data exchange between a system code, GENFLO, and FRAPTRAN is illustrated in Figure 5. At present, the boundary conditions for GENFLO from the system code are the mass flow and enthalpy at the channel inlet, the pressure at the top of the channel, and the total power and power profile of the fuel rod.

The GENFLO model is being evaluated to supplement the selection of transient heat transfer models available to FRAPTRAN. Although GENFLO has been tested in different surroundings,a the coupled FRAPTRAN-GENFLO code will need some verification, including comparisons against the existing thermal-hydraulic models used by FRAPTRAN.

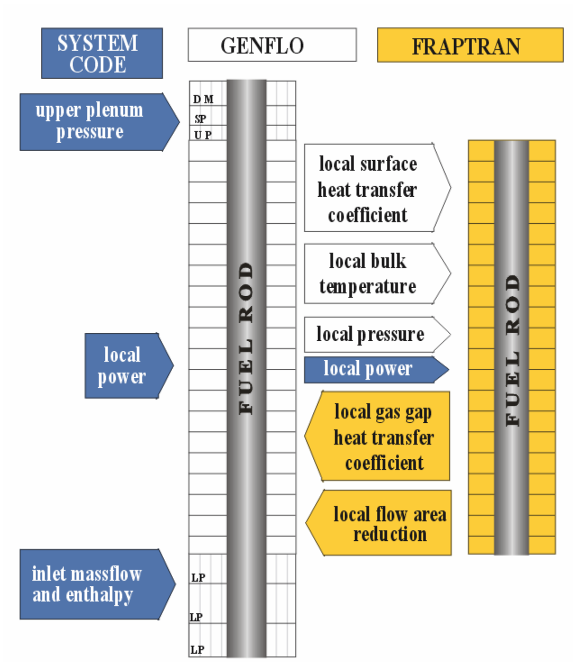

FIGURE 5. Data exchange between the system code, GENFLO and FRAPTRAN.

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a The GENFLO code is being used in two other applications in addition to FRAPTRAN. First, in the code RECRIT (Miettinen et al. 2000), GENFLO is coupled with the two-dimensional transient neutronics model TWODIN. RECRIT has been used for analysing recriticality incidents for BWRs under conditions where the control rods have melted at high temperatures but after core cooling has been recovered. Because the thermal-hydraulic solution of GENFLO has been included in the RECRIT code, the validation of RECRIT also supports the GENFLO thermal-hydraulics. The validation cases for the RECRIT code include ERSEC, ACHILLES, REWET-II, GÖTA, FLECHT and QUENCH experiments. For this case, a whole BWR vessel has been modelled for GENFLO. Second, in the APROS-SA application, GENFLO is used to calculate the PWR pressure vessel thermal-hydraulics during a severe accident until core melting, relocation, and pool generation at the bottom of the reactor vessel is simulated.

TWO EXAMPLE CALCULATIONS USING FRAPTRAN-GENFLO

The first example using FRAPTRAN-GENFLO is for fuel behaviour during a hypothetical large break LOCA (LBLOCA) at the Loviisa VVER-440 power plant. The engineering simulator APROS has been used to generate LBLOCA analyses. The thermal-hydraulic boundary conditions, as well as geometry and safety factors used in hot channel analyses of the LBLOCA have been averaged for one sub-channel and then used as input for FRAPTRAN-GENFLO. The initiating event for the transient is a double-ended break between the pressure vessel and the reactor coolant pump in one of the six cold legs. The LBLOCA transient is assumed to occur at beginning-of-life when the fuel has no significant burnup. As a comparison to the effects of using GENFLO thermal-hydraulics with FRAPTRAN, the same case was calculated by FRAPTRAN using a simple built-in coolant model (“separate” calculation).

In the transient, the fuel rod temperatures increase in the nearly dry core until injection from the emergency core cooling system quenches the core. Besides the high and low pressure safety injection, as simulated in the APROS calculation, two hydro accumulators start an injection to the downcomer. The other two accumulators, injecting to the upper plenum, are not operating in this transient.

A reactor trip immediately follows the pipe break. Due to the large break flow, coolant pressure decreases quickly and the core is uncovered. In this situation, the small flow to the hot channel is voided and is occasionally reversed. The real physical oscillation of inlet flow is difficult to separate from the numerical one calculated by APROS. To eliminate the fastest numerical frequencies, the channel inlet mass flow and enthalpy results have been filtered to provide a simpler history for GENFLO. The effects of filtering are seen in differences in the inlet liquid mass flow in Figure 6 between the GENFLO and APROS curves. Another reason for using filtering was the decision to calculate the same transient with the standard FRAPTRAN, which does not accept reversed flow.[b]

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b The coolant enthalpy model of FRAPTRAN was considered able to give the best possibility for a comparison with results not strongly dependent on the properties of an external thermal-hydraulic code. The raw transient data has to be slightly modified by filtering in order to avoid possible instantaneous coolant reverse flow, which could not be handled by the coolant enthalpy model. Hence, the comparison case is possibly not completely realistic in all details, but it is a test and an indication of the capabilities of the models and also serves as a tool for revealing deficiencies or necessary improvements in the models.

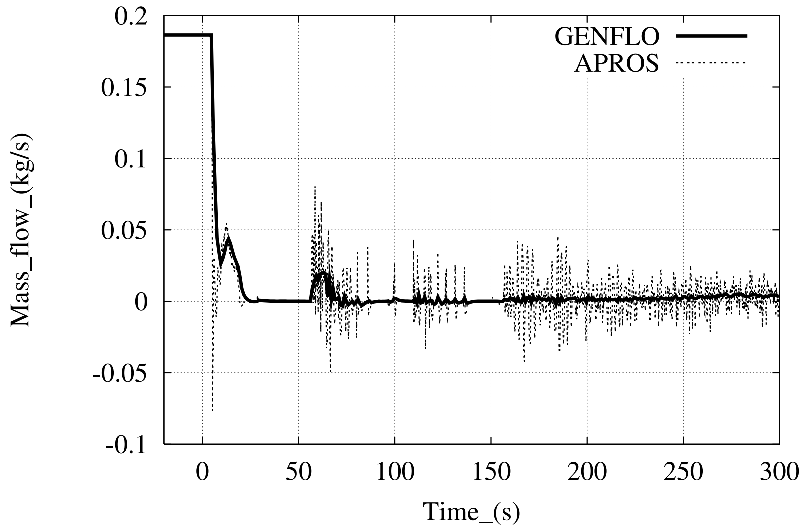

FIGURE 6. Inlet liquid flow at bottom of channel for LOCA example

At the beginning of the transient, the fuel temperature quickly drops because of the power decrease, as shown in Figure 7 (0-10s). Later, due to the loss of coolant inventory, the fuel temperature starts increasing (~25s). The coupled code suggests that even a small amount of water in the channel inlet after 60s is sufficient to first stop the temperature increase and then to temporarily decrease the temperatures.

At about 150s into the transient, the average void fraction is at its maximum with the water level at its minimum. The fuel temperatures increase and the quench front of the hydraulic channel drops still until 200s at which time the quench front starts to rise again. At the time of maximum temperature, there is a quite sharp transition from one flow regime to another.

The FRAPTRAN-GENFLO and FRAPTRAN calculations assume similar enthalpy values, even though for the separate FRAPTRAN calculation the enthalpy of the inlet flow may be unrealistic at the time of flow reversal.

The results from the separate FRAPTRAN and FRAPTRAN-GENFLO calculations are quite similar until the steam supply starts. The bulk temperature significantly increases due to high enthalpy values in the separate FRAPTRAN calculation. There are also corresponding differences in the heat transfer coefficients, rod temperatures and cladding deformation. Used separately, FRAPTRAN predicts that the clad starts ballooning at the time of the highest cladding temperatures. This is shown in Figure 8 by the decreasing plenum gas pressure at 150s. The internal pressure of the rod is seen to undergo rapid changes and then equals the channel pressure when the rod fails at about 250s (Figure 8). In contrast, no cladding ballooning or rod failure is predicted to occur by FRAPTRAN-GENFLO.

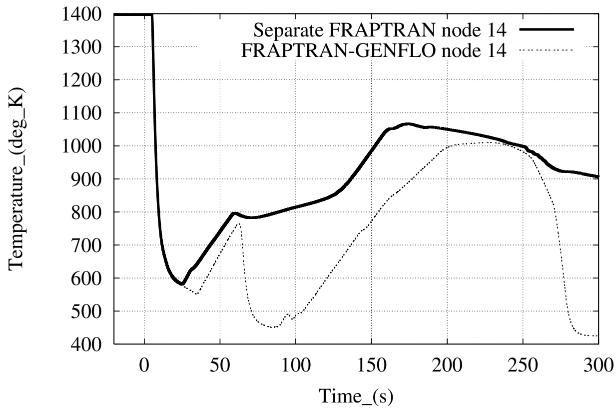

FIGURE 7. Fuel centerline temperature at level 14 / 20.

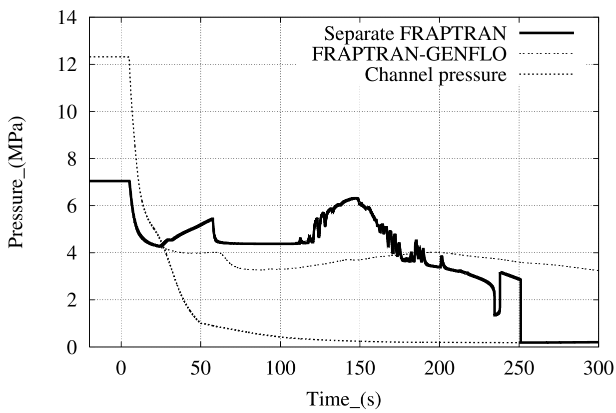

FIGURE 8. Channel pressure and rod plenum gas pressure

The second example calculation for FRAPTRAN-GENFLO is for an ATWS at a BWR plant. The basis for the analysis is an oscillation incident in the Finnish Olkiluoto 1 BWR during reactor startup on February 22, 1987. The incident was safely terminated by normal operation of the reactor safety systems.

The incident was analyzed by STUK (Valtonen 1989) and simulated with the Finnish TRAB-3D code (Daavittila et al. 2000). The results of the TRAB-3D calculation agree with measurements and earlier analyses. The oscillation frequency and the phase shift between the inlet and outlet flows in a channel of high relative power show good consistency. So do the out-of-phase oscillation of mass flows between high power channels and the core by-pass channel.

To test the performance of the new model combination, the case was also hypothetically extended assuming no actions of the safety system. The transient was recalculated with TRAB-3D as an ATWS case. The escalating oscillation phase of this calculation was chosen a subject of further studies in this FRAPTRAN-GENFLO analysis. The oscillations of boundary conditions were artificially continued in time and amplified. The total power in the hot assembly calculations with TRAB was multiplied by a factor of 1.3 for the hot rod analysis with FRAPTRAN-GENFLO. Also, the oscillating power was given a minimum value. Shown in Figure 9 is the boundary condition of the channel inlet mass flow. The flow rate oscillation finally leads to temporary flow reversals (negative mass flow).

A continuously changing axial power profile was created to match the TRAB-3D calculations. The axial power profile was dynamically changed in GENFLO. For FRAPTRAN, an average profile was used. Contrary to the original TRAB-3D calculation, high burnup fuel was assumed with a value of 62.3 MWd/kgU. The frequencies of oscillations of boundary conditions were according to the TRAB-3D calculations.

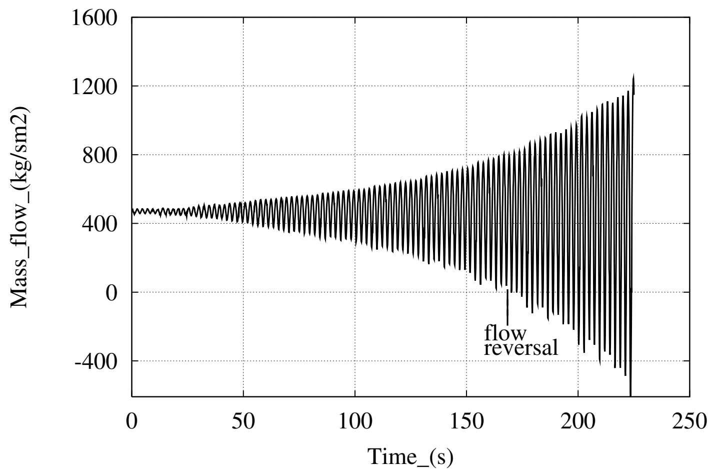

FIGURE 9. The mass flow rate boundary condition at channel inlet in hypothetical BWR instability case

The results of the FRAPTRAN-GENFLO calculations for this BWR instability case show that with a power cycle period of about 2s the fuel rod remains covered with water until the time of flow reversal. Then the quench front starts dropping in the channel. Before the flow reversal, only local or temporary dryout or DNB conditions may be achieved. The flow reversal soon leads to high cladding temperatures at the upper part of the fuel rod (1682K). On the other hand, the highest fuel temperatures occur at the lower part of the rod (2729K), where the linear power is higher. The temperature profiles are shown in Figures 10, 11, and 12. At the end of the calculation the cladding is quite soft and in contact with the fuel pellet (PCMI, pellet-to-cladding interaction). The plenum gas pressure remains below the fluid channel pressure during the whole transient, and no rod failure is predicted in spite of the fast deformation of the fuel rod (Figure 13).

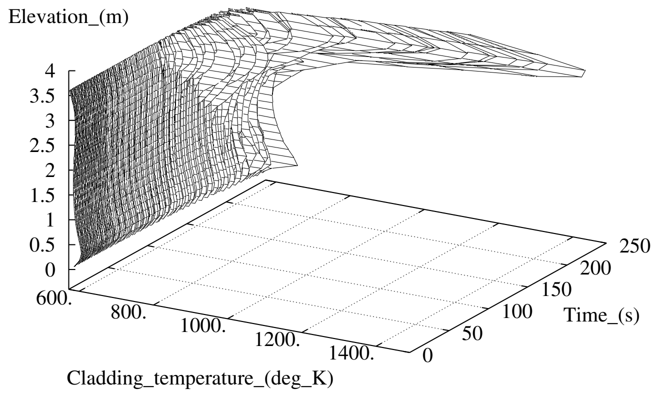

FIGURE 10. Cladding surface temperature profile in hypothetical BWR instability case

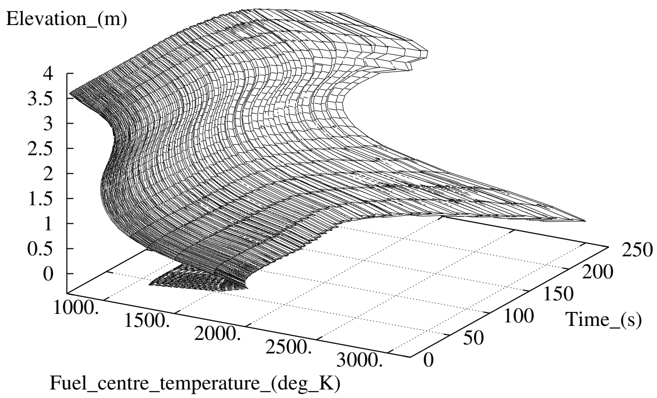

FIGURE 11. Fuel centre temperature profile in hypothetical BWR instability case

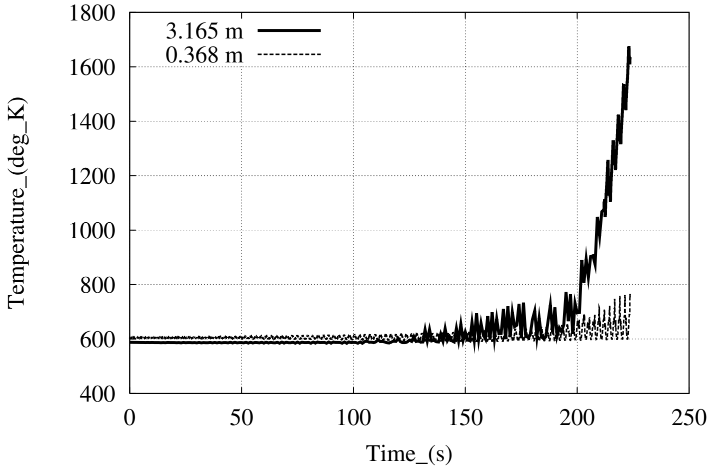

FIGURE 12. Cladding temperature at two axial level in BWR instability case

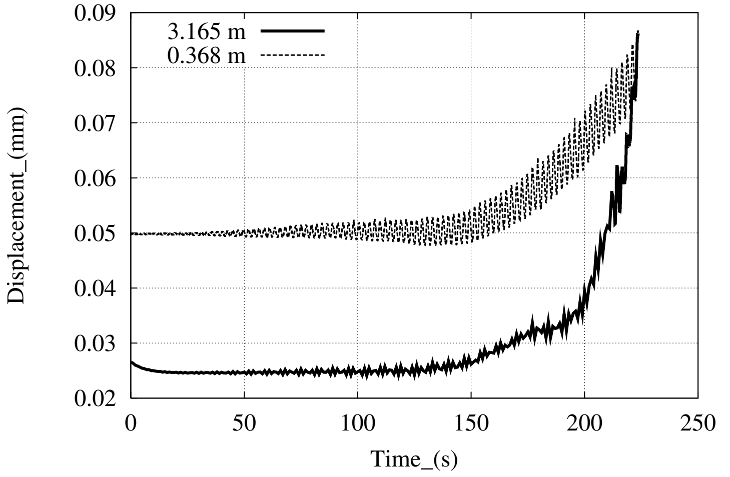

FIGURE 13. Displacement of cladding outer surface at two axial level in BWR instability case

FRAPTRAN-GENFLO is proving to be a proper tool for studying oscillation phenomena in a single subchannel, although continuation of the oscillation in real geometry may be very different from the oscillations assumed for this example. Another result from this study is that the critical heat flux correlations in GENFLO need further review.

DISCUSSION

The coupling of the FRAPTRAN and GENFLO codes demonstrates a direction to proceed with more detailed thermal-hydraulic description in fuel transient analyses. The results of the two examples illustrate that the coupling is successful and can result in different predicted fuel behavior than using FRAPTRAN alone. To build on this preliminary work, modifications and improvements to both codes have been identified and are being planned.

In order to benefit from the more detailed thermal-hydraulic input available from GENFLO, FRAPTRAN needs modifications to enable it to take advantage of accepting dynamic axial power and pressure profiles. Also, an increasing number of the system code calculations are now providing dynamic power profiles from 1 and 3-dimensional neutronics calculations in the core.

Some features already present in GENFLO are not enabled in the FRAPTRAN-GENFLO application. Examples include the flexible hydraulic channel geometry in GENFLO due to ballooning and an option to use more than one subchannel in a single run. Using several channels with different flow areas could describe the different positions of a fuel rod in an assembly. Further, some GENFLO models, like departure from nucleate boiling (DNB) and critical power ratio (CPR) correlations, that are suitable for a single hot rod and for BWR instability cases, need additional attention. Presently when using GENFLO, it is necessary to use the material properties within GENFLO. For consistency, the same material properties, valid to high burnups levels, should be used in both codes.

SUMMARY AND CONCLUSIONS

The FRAPTRAN transient fuel analysis code has been developed by the NRC and is now being made available to the FRAPCON-3 users group. This code has been assessed against a data base that emphasized experiments investigating the effect of burnup on fuel rod behavior during RIA and LOCA transients. FRAPTRAN has been shown to generally perform well in the comparisons to data. An independent peer review of FRAPTRAN also concluded that FRAPTRAN was able to predict the trends in the experimental data and that the code has been assessed against data appropriate for the intended uses of the code.

GENFLO, a thermal-hydraulic subchannel code, has been developed by VTT and has been coupled with FRAPTRAN. Two example cases, a Loviisa LBLOCA and a BWR instability case, have shown that coupling FRAPTRAN with GENFLO can work successfully. The coupling of the codes enables the evaluation and studying of the improvements and changes necessary in both the FRAPTRAN and GENFLO codes, and further features may be inserted easily. Several improvements for both codes have been introduced.

For the LBLOCA case, no cladding ballooning was predicted with the coupled code. In contrast, the separate FRAPTRAN calculation predicted cladding ballooning and rod failure because of probably unrealistic boundary conditions. For the BWR instability case, the calculation shows that with small cycle times the channel is wetted and rod failure does not occur until flow reversal. Definitive analyses need more carefully studied input and boundary conditions for both cases.

The results of the example cases show that GENFLO can be used as a thermal-hydraulic model for FRAPTRAN. From these two cases, it may be concluded that FRAPTRAN-GENFLO may be effectively used for single fuel rod and subchannel analyses. The results of this first version of the coupled code are encouraging and encourage further development in conjunction with careful validation against experimental data.

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Published by

Series title, number and report code of publication

VTT Publications 578
VTT–PUBS–578

Author(s)
Hämäläinen, Anitta

Title

Applying thermal hydraulics modeling in coupled processes of nuclear power plants

Abstract

This thesis focuses on the validation of the coupled codes developed in Finland for the safety analyses of the light water reactors in design basis accidents. The validation efforts and applications of the thermal hydraulics code SMABRE and the three-dimensional neutron kinetics codes are introduced for both the separate and coupled codes. The code development and coupling of codes for the safety analyses in Finland are discussed, and the present situation and possible future directions are described.

The data for the code validation consists of the experimental data from test facilities, numerical benchmarks and data measured in real nuclear power plants. The nuclear core is relevant for the couplings of neutron kinetics and thermal hydraulics codes. Two European Union projects, in which ten transients at real VVER plants have been documented, as well as the other international benchmarks are dealt with as useful forums in the code validation.

The simulation of plant measurements and several plant modeling aspects emerging from the validation work are gathered together. The main steam line break in different kinds of plants is dealt as an example of the application of the coupled codes in safety analysis. The coupling of a thermal hydraulic code with a fuel transient performance code is illustrated in the thesis as a new approach to performing the safety analyses. Another new approach is the sensitivity and uncertainty analyses performed for a Loviisa plant turbine trip case.

In addition to this summary, the thesis consists of seven publications in appendices: five articles in scientific journals and two conference papers.

Keywords
thermal hydraulics, modeling, coupled processes, nuclear power plants, code validation, coupled codes, safety analysis, neutron kinetics, SMABRE, HEXTRAN

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ISBN	Project number
951–38–6667–X (soft back ed.) 951–38–6668–8 (URL:http://www.inf.vtt.fi/pdf/)	C3SU00180

Date	Language	Pages	Price
October 2005	English	103 p. +app. 99 p.	E

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VTT Publications 1235–0621 (soft back ed.) 1455–0849 (URL: http://www.vtt.fi/inf/pdf/)	VTT Information Service P.O.Box 2000, FI–02044 VTT, Finland Phone internat. +358 20 722 4404 Fax +358 20 722 4374

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ISBN 951-38-6667-X (soft back ed.)
ISSN 1235-0621 (soft back ed.)

ISBN 951-38-6668-8 (URL: http://www.vtt.fi/inf/pdf/)
ISSN 1455-0849 (URL: http://www.vtt.fi/inf/pdf/)

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Anitta Hämäläinen