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Nuclear reactor core modelling in multifunctional simulators

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VTT Energy

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Abstract

This thesis concentrates on the development of nuclear reactor core models for the APROS multifunctional simulation environment and use of the core models in various kinds of applications. The work was started in 1986 as a part of the development of the entire APROS simulation system. The aim was to create core models that would serve in a reliable manner in an interactive, modular and multifunctional simulator/plant analyser environment. One-dimensional and three-dimensional core neutronics models have been developed. Both models have two energy groups and six delayed neutron groups. The three-dimensional finite difference type core model is able to describe both BWR- and PWR -type cores with quadratic fuel assemblies and VVER-type cores with hexagonal fuel assemblies. The one- and three-dimensional core neutronics models can be connected with the homogeneous, the five-equation or the six-equation thermal hydraulic models of APROS.

The key feature of APROS is that the same physical models can be used in various applications. The nuclear reactor core models of APROS have been built in such a manner that the same models can be used in simulator and plant analyser applications, as well as in safety analysis. In the APROS environment the user can select the number of flow channels in the three-dimensional reactor core and either the homogeneous, the five- or the six-equation thermal hydraulic model for these channels. The thermal hydraulic model and the number of flow channels have a decisive effect on the calculation time of the three-dimensional core model and thus, at present, these particular selections make the major difference between a safety analysis core model and a training simulator core model.

The emphasis on this thesis is on the three-dimensional core model and its capability to analyse symmetric and asymmetric events in the core. The factors affecting the calculation times of various three-dimensional BWR, PWR and VVER-type APROS core models have been studied to assess the possibilities for using three-dimensional cores in training simulators. The core model results have been compared with the Loviisa VVER-type plant measurement data in steady state and in some transients. Hypothetical control rod withdrawal, ejection and boron dilution transients have been calculated with various three-dimensional core models for the Loviisa VVER-440 core. Several ATWS analyses for the VVER-1000/91 plant have been performed using the three-dimensional core model. In this context, the results of APROS have been compared in detail with the results of the HEXTRAN code. The three-dimensional Olkiluoto BWR-type core model has been used for transient calculation and for severe accident re-criticality studies. The one-dimensional core model is at present used in several plant analyser and training simulator applications and it has been used extensively for safety analyses in the Loviisa VVER-440 plant modernisation project.

Preface

This work has been carried out at VTT Energy during the development of the APROS Simulation Environment by VTT Energy, VTT Automation and IVO Power Engineering Ltd (IVO PE)[1].

I thank my supervisor, Prof Rainer Salomaa, for his interest and encouragement in the process of preparation of this thesis.

I want to express my thanks to all co-authors of the publications in this thesis. I also want to thank all those persons of VTT and IVO with whom I have had the pleasure to work during the sometimes troubled twelve years of the APROS development.

In particular I want to thank my colleagues of the APROS development staff Mr Markku Hänninen and Mr Jukka Ylijoki from VTT Energy and Mr Harri Kontio and Mr Kari Porkholm from IVO PE. I am also grateful to Mr Kalle Kondelin from VTT Automation and Mr Jarto Niemi from VTT Energy for helping me through many practical problems of APROS. I want to thank also Mr Sixten Norrman and Mr Jaakko Miettinen from VTT Energy for good cooperation in many tough applications.

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Espoo, May 1998 Eija Karita Puska

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List of publications

I Puska, E. K., Hänninen, M. and Porkholm, K. 1994. Reactor Core Analysis in the Modular Plant Analyser of Loviisa Nuclear Power Plant. Proceedings of the 1st JSME/ASME Joint International Conference on Nuclear Engineering, Tokyo, Japan, November 4–7, 1991. Vol. 2. Pp. 541–546.

II Puska, E. K., Hänninen, M., Ylijoki, J. and Porkholm, K. 1992. Modular Plant Analyser of an Entire Power Plant: Extent and Validation. Proceedings of the 1992 Topical Meeting on Advances in Reactor Physics, Charleston, SC, USA, March 8–11, 1992. Vol. 2. Pp. 152–163.

III Puska, E. K., Hänninen, M. J., Porkholm, K. J. and Kontio H. E. 1994. Assessment of APROS Against the Loviisa-2 Stuck-Open Turbine By-Pass Valve Transient. Proceedings of the 1994 ASME Power Plant Transient Symposium, Chicago, Illinois, USA, November 6–11, 1994. American Society of Mechanical Engineers, FED-Vol. 204. Pp. 31–42.

IV Puska, E. K. and Kontio, H. 1995. Three-Dimensional Core in APROS Plant Analyser. Proceedings of the International Conference on Mathematics and Computations, Reactor Physics, and Environmental Analyses, Portland, Oregon, USA, April 30–May 4, 1995. American Nuclear Society Inc. Volume I. Pp. 264–273.

V Puska, E. K., Miettinen, J., Hänninen, M., Kontio, H. and Honkoila, K. 1995. APROS Simulation System for Nuclear Power Plant Analysis. Proceedings of ICONE-3, The 3rd JSME/ASME Joint International Conference on Nuclear Engineering, Kyoto, Japan, April 23–27, 1995. The Japan Society of Mechanical Engineers. Volume 1. Pp. 83–88.

VI Puska, E. K., Norrman, S., Miettinen, J. and Hänninen, M. APROS BWR Model for Transient Analysis. CD-ROM Proceedings of ICONE-5, 5th International Conference on Nuclear Engineering, Paper ICONE5-2700, Nice, France, May 26–30, 1997. ASME 8 p.

VII Puska, E. K., Norrman, S., Kyrki-Rajamäki, R., Siltanen, P., Porkholm, K., Latyeva, L., Semishin V. and Kalugin, S. 1997. Advanced Process Simulation with APROS for VVER-91, The Next Generation Reactor Design for VVER-1000. Proceedings of the 1997 International Meeting on Advanced Reactor Safety (ARS`97), Orlando, Florida, USA, June 1–4, 1997. American Nuclear Society Inc. Volume II. Pp. 1122–1130. (Accepted for publication in Nuclear Safety Journal).

VIII Puska, E. K. and Ylijoki, J. 1997. Coupling of Neutronics and Thermal Hydraulics in APROS-3D Core Calculations. Proceedings of the Joint International Conference on Mathematical Methods and Supercomputing for Nuclear Applications (M&C+SNA´97), Saratoga Springs, New York, USA, October 5–10, 1997. American Nuclear Society Inc. Volume 2. Pp. 1391–1400.

The disputant has been the leading author of papers I–VIII. In paper I, the disputant has been responsible for the formulation and presentation of the reactor model equations and for the calculation of the core transient examples. In paper II, the disputant had an active role in performing the plant transient calculations and analysing their results. In paper III, the disputant created the three-dimensional core model for the plant and participated in the analysis of the results obtained. In paper IV, the disputant was responsible for the presentation of the reactor core model principles, creation of the three-dimensional core models described and performing the transient examples. In paper V, the disputant was responsible for the three-dimensional core model and transients performed using the model. In paper VI, the disputants main responsibility was creation of the three-dimensional core model and performing the transient described. The disputant also participated in the analysis of results obtained with the plant model. In paper VII, the disputant was responsible for the creation of the three-dimensional core model and had an active role in performing the calculations with APROS and analysis of the results. In paper VIII, the disputant was responsible for the description of the neutronics model and its connection with thermal hydraulics. The disputant created all the three-dimensional core models and performed the transients and calculation speed measurements and their analysis.

The disputants contribution to the one- and three-dimensional core models of APROS and their applications has been reported also in Refs. 9, 12–13, 15, 32–37, 46–48, 52, 54–66, 69–70 and 72.

Contents

Appendices of this publication are not included in the PDF version. Please order the printed version to get the complete publication (http://www.inf.vtt.fi/pdf/publications/1999/

List of symbols

t	is time, as subscript present time step
$t-1$	is time, as subscript previous time step
Δt	is time interval from t-1 to t
i	is subscript, 1 for fast flux and 2 for thermal flux
ϕ_1	is fast flux
ϕ_2	is thermal flux
v_1	is fast neutron velocity
v_2	is thermal neutron velocity
D_1	is fast diffusion coefficient
D_2	is thermal diffusion coefficient
Σ_{b1}	is sum of fast absorption cross section, removal cross section from fast to thermal group and the absorption coefficients of control rods and soluble poison
Σ_{b2}	is sum of thermal absorption cross section and the absorption coefficients of control rods and soluble poison plus xenon and samarium absorption
Σ_{12}	is removal cross section from fast to thermal group
β	is total fraction of delayed neutrons
β_j	is fraction of delayed neutron precursor group j (j=1...6)
λ_j	is decay constant of delayed neutron precursor group j (j=1...6)
C_j	is concentration of the j´th delayed neutron precursor group (j=1...6)
S_f	is fission source term
$\nu\Sigma_{f1}$	is fast fission product cross section
$\nu\Sigma_{f2}$	is thermal fission product cross section
l	is subscript for node to be calculated
m	is subscript for a neighbour node
n	is subscript for a neighbour node

1. Introduction

APROS (Advanced PROcess Simulator) is a multifunctional simulation environment for the dynamic simulation of nuclear and conventional power plant processes and for the simulation of industrial process dynamics. It has been developed by Technical Research Centre of Finland (VTT) and IVO Power Engineering Ltd (IVO PE)[2] since 1986 [1]. By the end of the year 1997 there had been sixteen software package deliveries for modelling and dynamic simulation of non-nuclear process plants to fifteen customers in nine countries. On the nuclear side there had been eight APROS software deliveries to seven customers in six countries. In addition, there have been some thirty consulting applications of various sizes in the area of process and control system engineering and twelve nuclear consulting applications of APROS simulation software for eleven customers in seven countries.

The APROS simulation environment consists of an executive system, model packages, equation solvers, a real-time database and interface models. The model packages containing the physical models and process components related to them are grouped into general and application specific packages [2, 3]. The key feature of APROS is that the same packages can be used in various simulation applications. In addition to the general packages, such as thermal hydraulics, each application needs an application specific package. Nuclear plant simulation requires the thermal hydraulic, process component, automation system and electrical system packages from the general library. In nuclear power plant modelling, the application specific package required is the nuclear component package, which contains the one- and three-dimensional reactor model components and pressurizer components.

In the APROS nuclear reactor core the one- or three-dimensional two energy group neutronics model is connected with the homogeneous, five- or six-equation thermal hydraulic model. The three-dimensional model is a finite-difference type model. An essential feature of APROS is the flexible combination of the neutronics with thermal hydraulic channels. APROS core

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models have been designed to be an integral part of the simulation environment with the same requirements of on-line calculation, interruption of simulation, modification of the model and continuation of the simulation with the modified model, as the other models of the simulation environment.

The strength of traditional engineering or training simulators is the detailed plant process and automation description and fidelity to plant response, whereas the strength of safety analysis programs is in the detailed physical models. The basic idea of APROS has been to combine these two. In APROS, the plant process and automation can be described with the same detail and using similar tools to those used in traditional engineering or training simulators. Simultaneously, the thermal hydraulic models of APROS are at the same level as those in the well-known thermal hydraulic analysis codes, like RELAP5 [4] or CATHARE [5], and there is also possibility for three-dimensional core description as in core dynamics codes like HEXTRAN [6], SIMULATE-3K [7] or TRAB-3D [8]. In APROS, a finite-difference type three-dimensional neutronics model is used, whereas in most other well-known three-dimensional core dynamics codes a nodal model is used. The finite-difference approach was selected for APROS, since the original goal was to develop a core model for plant analyser and training simulator applications.

There is some variation of the concept ´plant analyser´ in the literature. In the context of some codes, like RELAP5, the concept means just a combination of the original code and the graphics connected to it. In the context of some other codes, like APROS, the concept ´plant analyser´ means the combination of physical models, graphical user interface and output facilities that are a prerequisite for a system capable for nuclear power plant safety analysis. Thus, the plant analyser concept is considered to cover to a certain extent the engineering simulator, the training simulator and the safety analysis tool. When a detailed description of a plant is made for the purpose of plant design studies, it is called an engineering or design simulator. With the addition of operator displays and real time calculation capability the result is called a training simulator. When an accurate description of parts of the power plant process is made with emphasis on those components active in a set of specified transients and some conservativity assumptions are made both in the physical models and in the behaviour of the systems involved, the result is called a safety analysis tool.

The most extensive APROS nuclear plant applications realised so far are Loviisa VVER-440 plant analyser [9], Kola VVER-440 plant analyser [10] and compact training simulator [11], VVER-1000/91 plant analyser [Paper VII] and plant analyser of the Olkiluoto BWR-plant [Paper VI]. Three-dimensional core models have been included in Loviisa VVER-440, VVER-1000/91 and Olkiluoto BWR plant analysers. A three-dimensional APROS core model has been used in plant safety analysis for VVER-1000/91. Due to the increase of computer performance, creation of various symmetric three-dimensional core models with real time calculation capability is already possible, which extends the use of three-dimensional core models for training simulator applications, too. At the other extreme, APROS three-dimensional core model is at present used in severe accident re-criticality studies for a BWR core [12, 13]. A one-dimensional core model and an extension of the one-dimensional model, a three-dimensional neutron flux approximation model, are being used in CHASNUPP [14] full scope PWR-type training simulator. Both one- and three-dimensional core neutronics models are being used in full scope HAMMLAB 2000 BWR simulator [15]. APROS nuclear applications with one- or three-dimensional core models are run at present on various workstations, like those produced by Hewlett-Packard, DEC, Sun and Silicon Graphics.

The APROS multifunctional simulation environment has been presented in Chapter 2. A review of the role and requirements of reactor core models in various simulation applications is given and the major differences of APROS and other corresponding codes are discussed. The chapter also includes a short discussion of two major features of the APROS environment: interactivity and visualisation, and their effect on the reactor core models. Principles of APROS nuclear reactor core models including the basic equations, coupling of neutronics and thermal hydraulics, and core model construction have been presented in Chapter 3. Chapter 4 concentrates on the various three-dimensional BWR, PWR and VVER -type core models created with APROS and the calculation speed considerations with these core models. Chapter 5 presents applications of APROS three-dimensional core models in plant analysers, safety analysis, training simulators and in severe accident research. Chapter 5 also includes comparisons of APROS three-dimensional core model results with plant measurements and plant fuel management code results.

Paper I includes the principal description of the APROS one- and three-dimensional core models, and examples of early applications of the model. Paper II discusses the problematics of an entire modular plant analyser. Paper III covers the detailed study of the combination of three-dimensional core model and the alternative five- and six-equation thermal hydraulic models in a real plant transient. In addition, the paper discusses in detail the significance of complete plant modelling and modelling of the possible operator actions in order to reach the proper repetition of a real plant transient. Paper IV contains presentation of the three-dimensional core model neutronics equations and examples of the three-dimensional VVER core model in various transient types. Paper V presents the Loviisa plant analyser application with three-dimensional core model. Paper VI presents the APROS BWR plant analyser applications with one- and three-dimensional core models and discusses the effects of the five- and six-equation thermal hydraulic models in various transients. In Paper VII, the application of APROS three-dimensional core model together with the plant process and automation model for VVER-1000/91 reactor is presented. The paper includes a comparison of the APROS results with the corresponding results of the HEXTRAN code in a main steam line break transient. Paper VIII contains a description of the various alternatives in the coupling of neutronics and thermal hydraulics in the APROS three-dimensional core model and calculation speed considerations for various reactor types and coupling alternatives.

2. APROS Multifunctional Simulation Environment

2.1 Introduction

The one- and three-dimensional nuclear reactor core models are a part of the nuclear reactor model package in APROS multifunctional simulation environment. In this chapter the general features of this environment are first presented. APROS nuclear reactor core models are being used in engineering simulators, safety analysis, training simulators and in severe accident studies. The role of the reactor core model in these various application areas and the special requirements set by each application type to the reactor core model are discussed. The chapter ends with a short description of the special features required from the reactor core models due to the interactivity of the simulation environment, and a means for visualizing the structure and calculation results of reactor core models.

2.2 Simulation environment

The APROS multifunctional simulation environment has been developed and is constantly used for the dynamic simulation of conventional and nuclear power plant processes and for the simulation of industrial process dynamics. At present, the major part of APROS power plant applications are on the non-nuclear side. An APROS simulation environment consists of an executive system, model packages, equation solvers, a real-time database and interface models. The executive system, the equation solver, the communication interface systems and the simulation database are considered as the primary systems of APROS in the sense that other APROS systems largely depend on the services provided by the primary systems [16]. The executive system interprets the commands given by the user and manages the database and controls the simulation. The equation solver provides efficient tools for solving systems of linear equations. The APROS database enables fast data exchange between computerised models and provides data structures for convenient handling of the information associated with simulation experiments. The data in the database is organised as modules consisting of attributes. The values of the attributes are stored in variables that are used by calculation models and updated during the simulation. The modules in the database can be connected to each other by means of name references and hierarchial relationships. The contents of the database can be saved at any time into a so-called snapshot file that can be reloaded and used as starting point of further simulation experiments. The communication interface systems administer all communications with other processes, like graphical user interfaces.

APROS software consists of physical models that are grouped into general and application specific packages. The general packages include thermal hydraulics, automation systems and electrical systems. The application specific packages include nuclear reactors, chemical reactions, boiler models, paper mill components, as well as diesel motor and gas turbine models. With the combination of various general packages, like thermal hydraulics or automation systems, and application specific packages, like nuclear reactors, boilers or paper mill components, various simulation applications in the field of nuclear or conventional power plant simulation or process industry can be created. The packages required for nuclear applications are the nuclear reactors, thermal hydraulics, automation and electrical systems.

APROS has been programmed using Fortran 77 and C languages. The code has been entirely written from scratch. However, in each application area, the developers have tried to benefit from all existing knowledge and experience gained in other development projects or assignments. The fact that the general software packages, like thermal hydraulics, as well as the data base structure and graphical user interface are used both in nuclear and in conventional plant applications, increases the reliability of these parts of the APROS system considerably by increasing the amount of users and resources available for further development in comparison with those possible within the scope of the nuclear applications only.

Figure 1 indicates various possibilities to use the APROS simulation environment in nuclear applications starting from the design of a new plant and ending in the use of APROS during the plant lifetime for training, planning of plant design changes and update of plant safety analysis [2]. One- and three-dimensional core models can be used in all these applications.

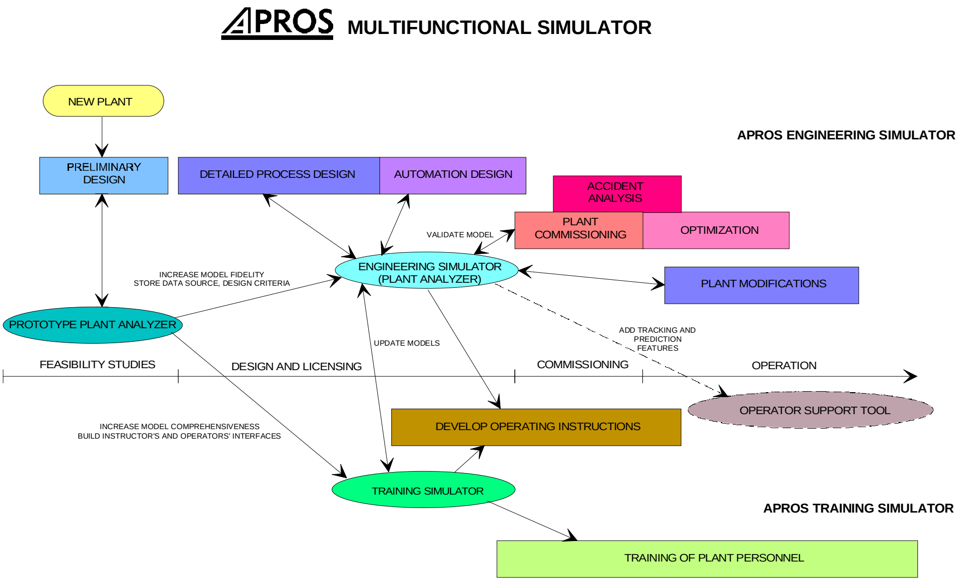

Figure 1. Use of APROS for engineering and training simulator applications and safety analysis during plant lifetime.

2.3 Role and requirements of reactor core models

Nuclear applications of APROS include, at present, engineering simulators or plant analysers, training simulators, plant safety analysis and severe accident research. The role and requirements of a reactor core model in these application areas may vary greatly. In some applications, the reactor must produce the proper power for the rest of the system and in some other applications the physical behaviour of the reactor core is of primary interest. However, in each specific area it is required that the reactor core model must be an integral and modular part of the simulation environment with the same requirements of interactivity and possibility for modifications as all the other models.

An *engineering simulator or plant analyser* is used, for instance, when planning replacement of some part of an existing power plant. In this application the detailed and true description of those parts concerned in the replacement process is essential. For operations affecting the core, this requires the description of the core with the level of detail available in core input data axially, according to the available burn-up and enrichment distribution and describing each fuel assembly separately in radial direction. In axial direction, the number of nodes usually extends from 10 to 30. Depending on the reactor type, the typical number of assemblies varies from some 150 to 700. For the description of asymmetric transients, each fuel assembly must be placed in a separate thermal hydraulic channel. The proper connection of the core to the process and automation system is essential, too. In a plant analyser the choice of five-equation thermal hydraulic model and application of parallel core and process calculation on two multiprocessor computers increases the calculation speed.

In *training simulators*, the essential requirements are the real-time calculation capability and the fact that an experienced operator should not be able to detect any difference between the actual plant and the simulator behaviour in the normal operation or transient situations. The real-time requirement is no problem when a one-dimensional core model is used. With present workstations the real-time requirement can already be met with some three-dimensional VVER-type cores, whereas in BWR-type cores this requires lumping of several fuel assemblies into so-called macroelements and/or lumping of several fuel assemblies into the same thermal hydraulic channel. In training simulators, fairly mild transient cases are involved. The requirement of fidelity of the simulator behaviour with the real plant behaviour is well ensured with the physical models and detailed plant description allowed in APROS.

In *safety analysis*, the most essential feature is the reliable result that must be calculated taking into account all the required conservativity factors in the analysis. Thus, for the description of asymmetric core transients originating either from events in the core, like control rod ejection, or outside the core, like erroneous start of a main circulation pump, the detailed description of core with fuel assemblies placed in separate flow channels is essential. Safety analysis may also require the use of the six-equation thermal hydraulic model in some situations, like analysis of a large break Loss Of Coolant Accident (LOCA). A typical feature of a safety analysis model is that only those parts of the plant process and automation having an active role in the transient are described. Hot channel calculations are an integral part of safety analysis. The APROS code has the capability of both simultaneous and post analysis hot channel calculations with any number of hot channels. Safety analysis applications require good physical models, since the cases discussed can be of a very extreme nature. In safety analysis the calculation speed is not a primary parameter.

In the *severe accident research* area, it is planned to use APROS to develop and validate calculation tools needed to plan preventive measures and to train the personnel for severe accident mitigation. In severe accident analysis the primary criteria are physically plausible results. As a code originally designed for plant analyser and training simulator use, the capabilities of the present APROS in the area of severe accident studies are limited. Reactor core re-criticality studies are an area where there is some motivation to use the present code with its capability to combine three-dimensional core with plant process and containment model. The code applicability is expected to improve greatly within a few years with development of special severe accident models.

In the area of reactor models, the complete reactor dynamics calculation system created at VTT Energy [17] has been beneficial for APROS development. The TRAWA [18], TRAB [19] and BOREAS [20] codes of that system have served as the starting points of the APROS one- and three-dimensional core model development. APROS core models can use the reactor kinetics data created by this calculation system. For instance, in VVER-applications APROS can use the same reactor kinetics data as the HEXBU-3D [21] and HEXTRAN [6] codes. The three-dimensional dynamics code HEXTRAN has been especially useful in validation of the APROS three-dimensional VVER-type core model.

There are some major differences between APROS and other corresponding codes used in nuclear applications. The major difference between APROS and other well-known code systems combining a three-dimensional reactor core model with plant process description [22–27] is that in APROS the three-dimensional core model is an integral part of the simulation model, whereas the other approaches are based on combining reactor core dynamics codes and plant thermal hydraulics codes with various means of transferring data between the two, originally independent, code systems. In APROS, a finite-difference type three-dimensional neutronics model is used. In most other well-known three-dimensional reactor dynamics codes a nodal model is used. The finite-difference approach was selected for APROS, since the original goal was a training simulator type three-dimensional core. The finite-difference model was judged to be the most simple way to describe in a reliable manner the three-dimensional core behaviour resulting from events in the plant primary and secondary circuits. It was easy to program in a clear and modular manner, which was considered as an advantage for the code maintainability, and the philosophy and discretization method of the basic equations were compatible with the APROS modelling concept. Some comparisons of APROS core model results with plant measurement data and results of other 3-D core models have been presented in Chapter 5.

Apart from the recent use of PLIM [28] thermal hydraulics in the HEXTRAN and TRAB [22, 29] the core thermal hydraulic models in APROS are more advanced than in most other codes dedicated to reactor core dynamics. APROS core thermal hydraulic models are at the same level as the thermal hydraulic models in the well-known thermal hydraulic analysis codes, like RELAP5 [4] or CATHARE [5].

2.4 Interactivity and visualisation

A remarkable difference between APROS and other well-known codes, like RELAP5 [4], used for nuclear plant thermal hydraulic analysis, is the modularity and interactivity of the APROS concept. In APROS it is possible to make changes in the process structure during simulation and continue the simulation without the need of code recompilation.

The communication interfaces of APROS cover both the design interface that is used in process creation, modification and simulation and the presentation interface for the presentation of simulation results. For the design interface there are at present two alternatives, the old UNIX-based configuration tool GRINAP and a new Pentium/NT-based configuration tool GRADES with some presentation capability [15]. The main advantages of the new configuration tool include a common database for both simulator data and the configuration drawings assuring much better consistency between the model and the drawing than the older tool GRINAP. For the presentation of process displays and simulation results there are various tools in UNIX and NT environments, like for instance the PICASSO-3 system [30] used for the creation of the Kola training simulator displays or the GNUPLOT program [31] used for the presentation of APROS three-dimensional core model simulation results.

Use of the reactor models via the graphical user interface with the possibility to make changes either in the core model itself or in the process or automation system during simulation results in two requirements: the calculation models must have sound physical basis and simultaneously they must be robust enough in order to tolerate the eventual transients resulting from the sudden changes in the core or process. In the core, the user can make changes in the core general data, in the assembly and channel components and the user can also change fluently the division of the assemblies into thermal hydraulic channels. Figure 2 shows the presentation of a primary circuit model with pressure vessel with the UNIX-based GRINAP tool. By clicking the connection point in the pressure vessel the user can obtain the core presentation shown in Figure 3.

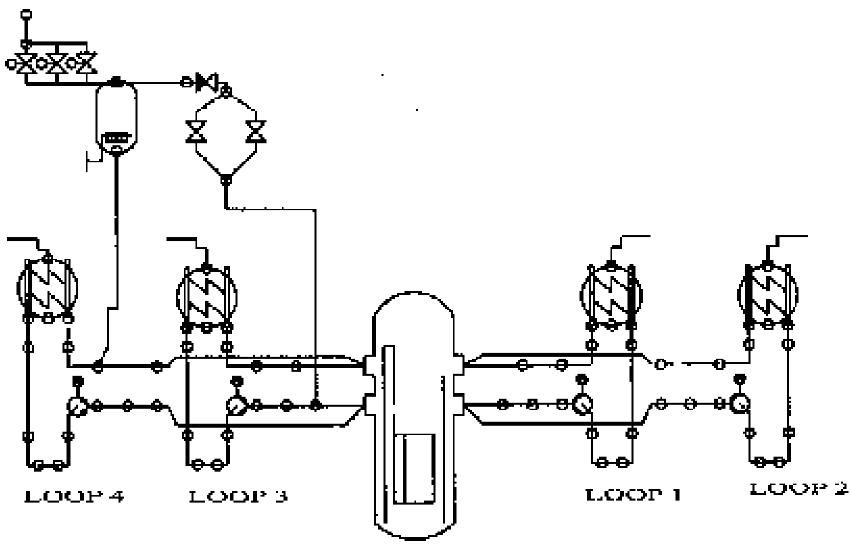

Figure 2. Presentation of primary circuit process and pressure vessel with core symbol on workstation screen.

The results of three-dimensional core calculations were presented in the early applications with histograms or core maps typically with three different colours presenting values in certain power, temperature or density ranges [32–33] on the screen of a separate PC. Later on presentation of calculated three-dimensional core results on-line from any axial level became possible on workstation screen [34–35]. At present, this type of presentation is created with the freely available GNUPLOT program that is provided together with APROS. Some examples of pictures plotted out using this type of presentation have been given in Chapter 5.

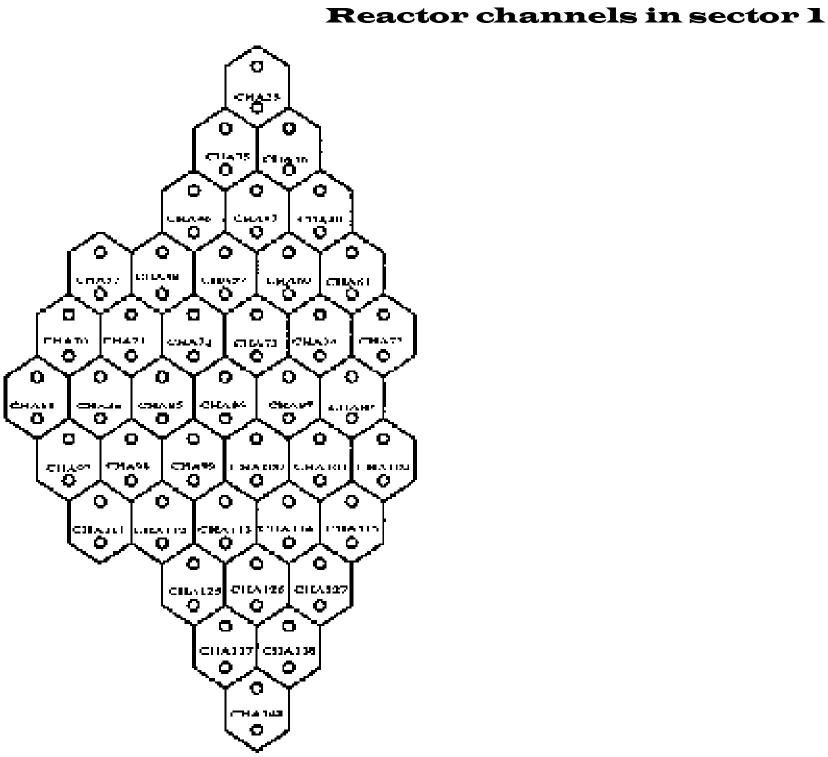

Figure 3. Presentation of a segment of three-dimensional core on workstation screen.

3. Core model principles

3.1 Governing equations

APROS code has one-dimensional and three-dimensional core neutronics models. Both models have two energy groups and six delayed neutron groups. In the models, the basic equations are first discretized. In the three-dimensional model, the neutron flux equations are integrated over the node volumes, a few approximations are made and the fast and thermal flux equations are solved using Gauss-Seidel iteration process. The finite-difference type three-dimensional neutronics model is able to describe both hexagonal and quadrilateral fuel assembly geometry [Papers I, IV, Ref. 36]. The concentrations of six delayed neutron precursor groups are calculated. Iodine, xenon, samarium and promethium calculations are included, too, with user-selected speedup factor. Reactivity feedback effects due to fuel temperature, coolant density and temperature, coolant void fraction, coolant boric acid concentration and control and scram rods are taken into account in the models.

The one- and three-dimensional neutronics models of APROS are based on the two energy group, six delayed neutron precursor group diffusion equations for fast flux ϕ_1 and thermal flux ϕ_2

$$\frac{\partial}{\partial t}\frac{1}{\mathrm{v}_1}\phi_1 - \nabla \cdot D_1 \nabla \phi_1 + \Sigma_{b1}\phi_1 = (1-\beta)S_f + \sum_j \lambda_j C_j \tag{1}$$

$$\frac{\partial}{\partial t}\frac{1}{\mathrm{v}_2}\phi_2 - \nabla \cdot D_2 \nabla \phi_2 + \Sigma_{b2}\phi_2 = \Sigma_{12}\phi_1 \ . \tag{2}$$

In equation (1), the fission source S_f is expressed as

$$S_f = v\Sigma_{f1}\phi_1 + v\Sigma_{f2}\phi_2 \ . \tag{3}$$

In the above equations, v_1 and v_2 are the fast and thermal velocities, D_1 and D_2 the fast and thermal diffusion coefficients, Σ_{b1} is the sum of the fast absorption cross section, the removal cross section from the fast to thermal groups and the absorption coefficients of control rods and soluble poison. Σ_{b2} is the sum of thermal absorption cross section and the absorption coefficients of control rods and soluble poison plus xenon and samarium absorption. Σ_{12} is the removal cross section from the fast to thermal group. β is the total fraction of delayed neutrons. λ_j is the decay constant of the delayed neutron precursor group j and C_j is the concentration of the j´th delayed neutron precursor group. $\nu\Sigma_{f1}$ and $\nu\Sigma_{f2}$ are the fast and thermal fission production cross sections, respectively.

In the solution, the first approximation for the fast and thermal group equations, i = 1, 2, respectively, is [Paper I]

$$\frac{\partial}{\partial t}\frac{1}{\mathrm{v}_i}\phi_i = \frac{1}{\mathrm{v}_i}\frac{\partial}{\partial t}\phi_i \ . \tag{4}$$

The time derivative is written as

$$\frac{\partial}{\partial t}\phi_i = \frac{\phi_{i,t} - \phi_{i,t-1}}{\Delta t} \tag{5}$$

where Δt is the time interval from t-1 to t.

In the one-dimensional model, the next approximation is

$$-\nabla \cdot D_i \nabla \phi_i = -D_i \frac{\partial^2 \phi_i}{\partial z^2} \ ; \ i = 1, 2 \ . \tag{6}$$

Further, the second derivative is expressed in the discretized form

$$\frac{\partial^2 \phi_i}{\partial z^2} = \frac{\phi_{i,\ell+1} - 2\phi_{i,\ell} + \phi_{i,\ell-1}}{(\Delta z)^2} \ \ ; \ \ i = 1, 2 \ . \tag{7}$$

In the above equation, ℓ-1, ℓ and ℓ+1 refer to the successive nodes and equal mesh spacing Δz has been assumed for simplicity. However, the solution in the code takes into account the actual mesh spacing. In the equations of Chapter 3, the values of parameters are for the time step t when the time step subscript has not been indicated.

Taking into account equations (4), (6) and (7) and transferring the flux terms into the left hand side, a block-tridiagonal matrix equation of the form

$$\overline{A}\overline{\phi}_{\ell-1} + \overline{B}\overline{\phi}_{\ell} + \overline{C}\,\overline{\phi}_{\ell+1} = \overline{S}_{\ell}$$

is obtained for the one-dimensional model.

In the finite difference type three-dimensional APROS core neutronics model, the remaining equations are integrated over volume V_ℓ of a single node ℓ [Paper IV]

$$\begin{aligned}&\int_{V_\ell} \frac{1}{\mathrm{v}_1} \frac{\phi_{1,t} - \phi_{1,t-1}}{\Delta t} dV - \int_{V_\ell} \nabla \cdot D_1 \nabla \phi_1 dV + \int_{V_\ell} \Sigma_{b1} \phi_1 \, dV = \\ &\int_{V_\ell} (1-\beta)\left(\nu \Sigma_{f1} \phi_1 + \nu \Sigma_{f2} \phi_2\right) dV + \int_{V_\ell} \sum_j \lambda_j C_j dV\end{aligned} \tag{8}$$

$$\int_{V_\ell} \frac{1}{\mathrm{v}_2} \frac{\phi_{2,t} - \phi_{2,t-1}}{\Delta t} dV - \int_{V_\ell} \nabla \cdot D_2 \nabla \phi_2 dV + \int_{V_\ell} \Sigma_{b2} \phi_2 dV = \int_{V_\ell} \Sigma_{12} \phi_1 dV \ . \tag{9}$$

This results into

$$\left[\overline{\frac{1}{v_1}\frac{\phi_{1,t}-\phi_{1,t-1}}{\Delta t}}\right]V_\ell - \int_{V_\ell} \nabla \cdot D_1 \nabla \phi_1 dV + \left[\overline{\Sigma_{b1}\phi_1}\right]V_\ell =$$

$$\left[\overline{(1-\beta)(\nu\Sigma_{f1}\phi_1 + \nu\Sigma_{f2}\phi_2)}\right]V_\ell + \left[\overline{\sum_j \lambda_j C_j}\right]V_\ell \tag{10}$$

$$\left[\overline{\frac{1}{v_2}\frac{\phi_{2,t}-\phi_{2,t-1}}{\Delta t}}\right]V_\ell - \int_{V_\ell} \nabla \cdot D_2 \nabla \phi_2 dV + \left[\overline{\Sigma_{b2}\phi_2}\right]V_\ell = \left[\overline{\Sigma_{b2}\phi_2}\right]V_\ell \tag{11}$$

where the bar signifies average over volume V_ℓ, for instance

$$\left[\overline{\Sigma_{b1}\phi_1}\right]V_\ell = \frac{1}{V_\ell}\int_{V_\ell} \Sigma_{b1}\phi_1 dV \ . \tag{12}$$

The divergence theorem is used to transform the volume integral into surface integral over the bounding surface A_ℓ of the node V_ℓ. Thus the second term on the left hand side in equations (10) and (11) becomes

$$\int_{V_\ell} \nabla \cdot D_i \nabla \phi_i dV = \int_{A_\ell} D_i \bar{n} \cdot \nabla \phi_i dA \ . \tag{13}$$

The derivation of an expression for the surface integral in equation (13) is performed according to the same method as in the BOREAS code [20]. In the following, it is assumed for simplicity that node l and all its neighbour nodes m are identical in size and shape, and thus the distance between the centres of the two nodes, h_{lm}, is equal to the constant node width in the l-m -direction. In the actual program the fact, that the distance between the node centres can be different in various directions, has been taken into account. Let ϕ_{il} be the fast or thermal flux, i= 1 or 2, respectively, at the centre of the node l and ϕ_{im} be the corresponding flux at the centre of the node m and ϕ_{ilm} be the corresponding flux at the interface along the line connecting the node centres. Thus, the approximations used in the preceding equation are

$$-D_i \bar{n} \cdot \nabla \phi_i = -\frac{1}{2}\left(D_{im} \frac{\phi_{im} - \phi_{ilm}}{h_{lm}/2} + D_{il} \frac{\phi_{ilm} - \phi_{il}}{h_{lm}/2} \right). \tag{14}$$

According to the continuity condition at the interface, the following expression is obtained for the interface flux

$$\phi_{ilm} = \frac{D_{il}\phi_{il} + D_{im}\phi_{im}}{D_{il} + D_{im}} . \tag{15}$$

The surface integration in equation (13) corresponds to multiplying the equation by the factor

$$A_{lm} = V_\ell / h_{lm} = V_m / h_{lm} \tag{16}$$

for the quadrilateral lattice and by the factor

$$A_{lm} = 2V_\ell / 3h_{lm} = 2V_m / 3h_{lm} \tag{17}$$

for the hexagonal lattice.

In the quadrilateral lattice the same expression is, of course, used in all directions. In the hexagonal lattice, the expression for the quadrilateral lattice is used for the two vertical neighbour nodes (subscript n) and the expression for the hexagonal lattice is used for the six horizontal neighbour nodes (subscript m).

When the surface integration for the terms of equation (13) is performed, and the entire fast and thermal flux equations are divided by the common factor V_ℓ, and the terms containing ϕ_{il} are placed on the left hand side of the equations and the rest on the right hand side, the final equations obtained for the quadrilateral fuel assemblies are

$$\phi_{1l} = \left[\frac{1}{v_1 \Delta t} \phi_{1l,t-1} + \sum_n \frac{2D_{1l} D_{1n}}{h_{\ell n}^2 \left(D_{1l} + D_{1n}\right)} \phi_{1n} + (1-\beta)\nu \Sigma_{f2}\, \phi_{2l} + \sum_j \lambda_j C_j \right] /$$
$$\left[\frac{1}{v_1 \Delta t} + \sum_n \frac{2D_{1l} D_{1n}}{h_{\ell n}^2 \left(D_{1l} + D_{1n}\right)} + \Sigma_{b1} - (1-\beta)\nu \Sigma_{f1} \right] \tag{18}$$

$$\phi_{2l} = \left[\frac{1}{v_2 \Delta t} \phi_{2l,t-1} + \sum_n \frac{2D_{2l} D_{2n}}{h_{\ell n}^2 \left(D_{2l} + D_{2n}\right)} \phi_{2n} + \Sigma_{12}\, \phi_{1l} \right] /$$
$$\left[\frac{1}{v_2 \Delta t} + \sum_n \frac{2D_{2l} D_{2n}}{h_{\ell n}^2 \left(D_{2l} + D_{2n}\right)} + \Sigma_{b2} \right] \tag{19}$$

and correspondingly for the hexagonal fuel assemblies

$$\phi_{1l} = \left[\frac{1}{v_1 \Delta t} \phi_{1\ell,t-1} + \sum_m \frac{4D_{1l} D_{1m}}{3h_{\ell m}^2 \left(D_{1l} + D_{1m}\right)} \phi_{\ell m} + \sum_n \frac{2D_{1l} D_{1n}}{h_{\ell n}^2 \left(D_{1l} + D_{1n}\right)} \phi_{\ell n} \right.$$
$$\left. + (1-\hat{a})í \sum_{f2} \phi_{2\ell} + \sum_j \ddot{e}_j C_j \right] /$$
$$\left[\frac{1}{v_1 \Delta t} + \sum_m \frac{4D_{1l} D_{1m}}{3h_{\ell m}^2 \left(D_{1l} + D_{1m}\right)} + \sum_n \frac{2D_{1l} D_{1n}}{h_{\ell n}^2 \left(D_{1l} + D_{1n}\right)} + \Sigma_{b1} - (1-\beta)í\ \Sigma_{f1} \right] \tag{20}$$

$$\phi_{2l} = \left[\frac{1}{v_2 \Delta t} \phi_{2\ell,t-1} + \sum_m \frac{4D_{2l} D_{2m}}{3h_{\ell m}^2 \left(D_{2l} + D_{2m}\right)} \phi_{2m} + \sum_n \frac{2D_{2l} D_{2n}}{h_{\ell n}^2 \left(D_{2l} + D_{2n}\right)} \phi_{2n} - \Sigma_{12}\, \phi_{1\ell} \right] /$$
$$\left[\frac{1}{v_2 \Delta t} + \sum_m \frac{4D_{2l} D_{2m}}{3h_{\ell m}^2 \left(D_{2l} + D_{2m}\right)} + \sum_n \frac{2D_{2l} D_{2n}}{h_{\ell n}^2 \left(D_{2l} + D_{2n}\right)} + \Sigma_{b2} \right]. \tag{21}$$

The delayed neutron precursor concentration is obtained from the equation

$$\frac{\partial}{\partial t}C_j = \beta_j\left(\nu\Sigma_{f1}\phi_1 + \nu\Sigma_{f2}\phi_2\right) - \lambda_j C_j \tag{22}$$

using first-order discretisation for the time-derivative. In the above equation, β_j is fraction of delayed neutrons precursor group j.

There are several alternative options to calculate the iodine, xenon, promethium and samarium concentration in the code. The concentrations are assumed to be in equilibrium, or time dependent concentrations can be calculated. A speed-up factor can be used in the calculations. In the time-dependent calculations, the solution is obtained by using first-order discretisation of the time-derivative in the basic equations [37].

The nodal cross sections used in the above equations are obtained from nominal parameters and feedback data to account for the actual thermal hydraulic and other conditions of the nodes. The data and equations used depend on the application. For the VVER-reactor models the data and feedback correlations of the HEXBU-3D code have been used. In APROS code input, burn-up and enrichment are specified separately for each axial node of each fuel assembly, and in the above equations the two-group cross sections created by codes like CASMO [38] or CASMO-HEX [39] and condensed [40] for use in codes like APROS, HEXTRAN or TRAB-3D are used in each neutronics node.

In APROS there is also a so-called three-dimensional neutron flux approximation model. This model consists of time-independent solution of the two-energy group equations and a simplified solution of the five-equation thermal hydraulic model in a number of independent flow channels. This model works only in connection with the one-dimensional neutronics model and there is a strong feedback in the neutronics solution from the fast and thermal flux of the one-dimensional model. The model was created for the basic demonstration of three-dimensional core effects in cases where the computer capacity did not allow the use of the three-dimensional core model.

3.2 Coupling with thermal hydraulics

The one- and three-dimensional core neutronics models can be coupled with the homogeneous, the five-equation or the six-equation thermal hydraulic model of APROS. The homogeneous model [41] was originally used in core thermal hydraulics. At present, either the five-equation thermal hydraulic model [42] or the six-equation thermal hydraulic model [43] is used with the one- and three-dimensional neutronics models. The five-equation model is based on the conservation equations of mass and energy for liquid and gas phases and momentum equation for mixture of gas and liquid [42]. In the five-equation model, the gas and liquid interface friction is not calculated, but the differential phase velocities are obtained through the drift flux correlations. A separate drift flux model calculates the mass flow rates of the phases. The quantities to be solved in the model are pressures, volumetric flows, void fractions and phasial enthalpies. No iteration is needed in the model.

The six-equation model describes the behaviour of one-dimensional two-phase flow. The model is based on the conservation equations of mass, momentum and energy for the gas and liquid phases separately [43]. The equations are coupled with empirical correlations describing various two-phase phenomena. The pressures and velocities, volume fractions and enthalpies of each phase are solved from the discretized equations using an iterative procedure. A moving mesh model for axial heat conduction can be used if the heat flows have to be calculated with great accuracy.

Heat transfer modules connect the five- and six-equation thermal hydraulic models with their own heat conduction solutions. Boric acid concentration in the thermal hydraulic nodes is calculated using the concentration solver of the APROS system. The concentration solution is analog with the enthalpy solution in the six-equation thermal hydraulic model [43]. Boron is assumed to be transported from node to node only with the liquid phase. The thermal hydraulic part of APROS also contains calculation of fuel enthalpy and oxide layer thickness on cladding surface and power production by cladding oxidation according to Baker-Just model that are required for the hot channel calculations. The information from the core flow channels to the hot channels is transmitted with the boundary condition modules of APROS [43].

Calculation of fuel rod temperatures is performed in the thermal hydraulic part of APROS. The fuel rod is described as a solid heat structure consisting of three materials: fuel, fuel-cladding gap and cladding. One-dimensional heat conduction solution [41] in a fuel rod is calculated using ten radial nodes. Specific material properties can be given for the fuel pellet, gap and cladding without the need of re-compiling the code. A temperature dependent gap conductance calculation can be performed, too. At present, axial heat conduction in the fuel rod can also be taken into account.

Coupling of the neutronics and thermal hydraulics in the APROS core model at the process component level (see Chapter 3.3) is formed when the user constructing the core model indicates to which thermal hydraulic channel the fuel, reflector or control assembly belongs [Paper VIII]. Coupling of neutronics nodes with the thermal hydraulic and heat structure nodes of the thermal hydraulic model is performed at the code calculational level on the basis of the information given by the user at the higher process component level. The user can specify the minimum and maximum numbers of iteration cycles in neutronics and thermal hydraulics. Typically, in the three-dimensional neutronics model, two iteration cycles are used as minimum and ten to one hundred as maximum, and three iteration cycles as minimum in six-equation thermal hydraulics and ten to one hundred as maximum. In neutronics, the convergence is defined on the basis of fast flux and in the six-equation thermal hydraulic model the most important convergence criteria is the mass error calculated from mass balance.

In APROS, the thermal hydraulic model convergence is required to be reached first during each time step. Calculation of fuel temperatures, coolant densities and temperatures, void fractions and boric acid concentrations is performed in the thermal hydraulic part. This information is transferred to the neutronics model for use in the calculation of feedback corrections of nodal cross sections. The feedback correction calculation also requires information on the positions of control rods in the core, which is obtained from the automation models of APROS. Then the fast and thermal flux values and the six delayed neutron precursor group concentrations are calculated. The calculation is repeated until the flux values converge or maximum number of iteration rounds is reached. Based on the fast and thermal neutron flux values calculated for each neutronics node in the core and the decay heat model, the power produced in each node is calculated. The power produced is then converted into the relative power of each heat structure and transferred back to the heat structures of the reactor channels that are a part of the thermal hydraulic model. It is possible to direct part of the power produced directly to the thermal hydraulic node representing the coolant.

The five-equation thermal hydraulic model does not iterate, and thus after the calculation of the neutronics part the calculations for that specific time step are finished. When the six-equation thermal hydraulic model is used in the core the minimum number of iterations is usually three, and thus the relative powers of heat structures obtained as a result of the neutronics calculation are used in the thermal hydraulic part, the resulting fuel temperatures, coolant densities etc. are transferred to the neutronics part, and new relative power values are produced. This procedure is repeated until convergence in the thermal hydraulic part is reached.

There is also an automatic time step control in APROS. If convergence is not reached with the user-specified or default convergence criteria in thermal hydraulics with the allowable number of iterations, the time step size is halved and a new calculation starting from the values of the previous time step is initiated. The process is continued until convergence is reached or until the minimum allowable time step size is reached. With the same manner, after a few steps with the shortened time step size, the model starts to increase the step by doubling until the maximum time step length is reached.

The time step control in APROS was initially mastered solely on the basis of the convergence of the thermal hydraulic model. However, in nuclear applications there are also several situations, like reactivity peaks, that require a shortened time step, due to the power increase even, though the thermal hydraulic model would not indicate a need for time step reduction. If the relative fast flux change per time step is greater than allowed, the simulation time step will be reduced in the same manner as with the thermal hydraulic model [Paper VIII].

3.3 Core model construction

In APROS, the user operates with a graphical user interface using components at the process component level. These correspond typically to components that can be found in power plants, like pumps or pipes. The data required is similar to that found in the data sheets of the real plant components. The user can specify the connections of the process component into other process components, too. On the basis of the information given at the process component level, the system then creates the calculational level of nodes and branches. During the simulation, the user can change the process component data or change the connections of the process component to other process components without the need of re-compilation of the code.

The hierarchial structure of APROS and its analogy to the components of the reactor core model has been presented in Figure 4. In the three-dimensional core model the process components are the fuel assemblies, control assemblies, reflector assemblies and one-dimensional thermal hydraulic flow channels [Paper VIII]. For each fuel assembly, the user must specify a name, which usually corresponds to some identification code at the actual core; a x-y position in the core model; division of assembly in axial sections; burn-up; enrichment; and for BWR also void and control rod history in each axial section. Similar information plus indication of rod position is given for each control assembly. In a similar manner, reflector assemblies can be defined, or extrapolation lengths can be used instead. Additionally, for each fuel assembly and control assembly the name of the thermal hydraulic channel where it belongs must be specified. The introduction of the thermal-hydraulic channel concept [44] greatly improved the accuracy and applicability of the three-dimensional core model. For each thermal hydraulic channel, a name must be specified along with the type of thermal hydraulic model (five- or six-equation model), fuel rod geometry data, number of fuel rods, flow area of channel, hydraulic diameter, nominal power and division of the channel in the axial direction etc.

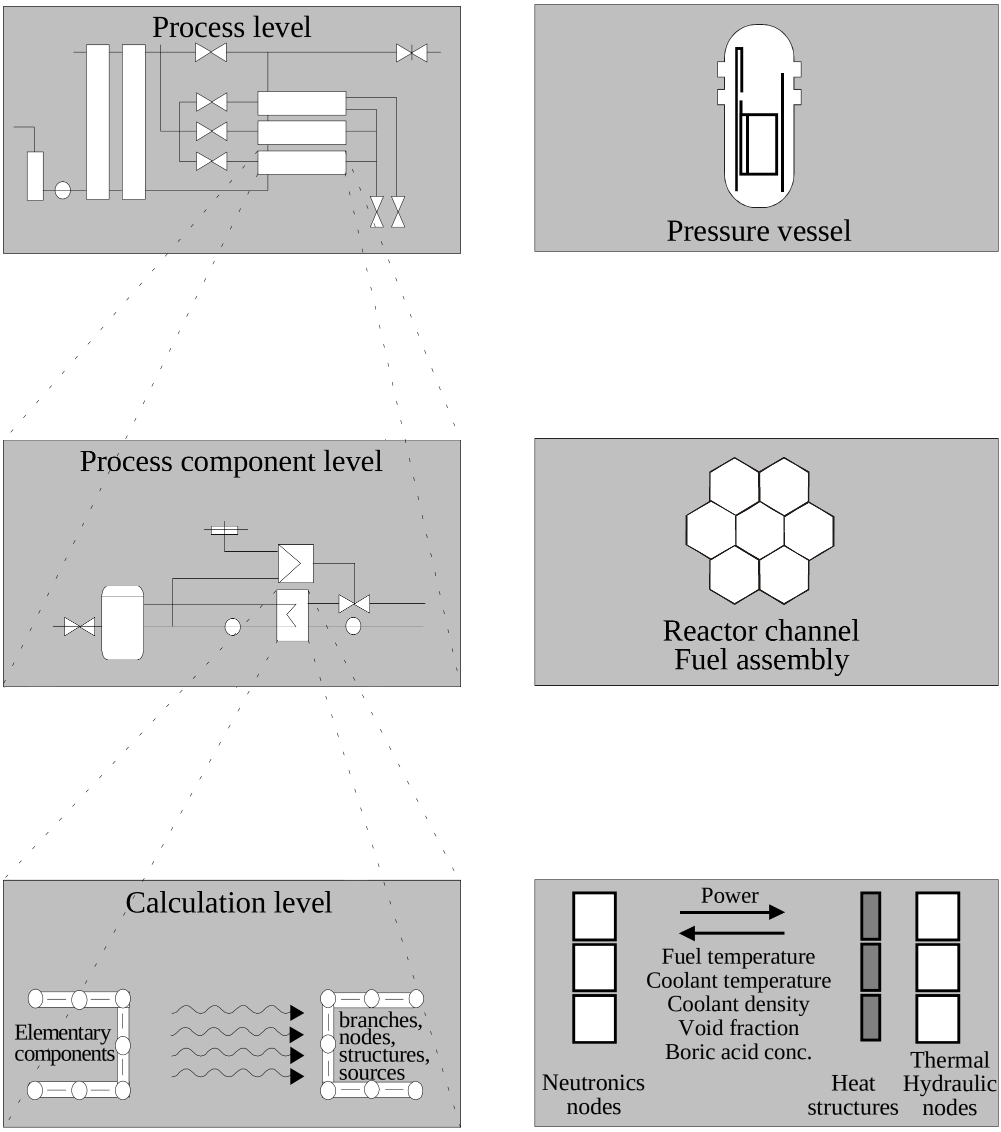

Figure 4. Hierarchial structure of APROS and its analogy to reactor core components.

In addition, the three-dimensional model requires some general information like indication of reactor type (BWR, PWR or VVER) and assembly geometry (hexagonal or quadratic) plus cross section data file. On the basis of the information given, the neutronics and thermal hydraulics calculational level descriptions are created by APROS. This also includes creation of the index tables telling which neutronics node is connected to which thermal hydraulic node in the core. All fuel and control assemblies are assumed to have the same size and axial division. These usually correspond to the real core fuel and control assemblies or in some cases to the so-called macro elements. In addition, the axial division is usually quite well defined by the nature of the core input data.

A specific feature of APROS is that the user can place the assemblies into thermal hydraulic channels according to the requirements of the calculation. In safety analysis, one fuel assembly per flow channel is used in cases where asymmetric behaviour in the core is expected and in symmetric cases identical assemblies are placed in the same channel. In real-time three-dimensional training simulator cores that require a description of asymmetric events, most of the core can be described with identical assemblies placed into same flow channels. The asymmetric part, like withdrawn assembly and the closest neighbours, are placed into separate flow channels.

4. APROS three-dimensional LWR core models

4.1 BWR, PWR and VVER core models

A number of APROS three-dimensional BWR, PWR and VVER core models have been created for various purposes. APROS three-dimensional core models have been applied for the safety analysis of VVER-91 type reactors for the calculation of various ATWS cases [Paper VII]. Full three-dimensional core models have also been created for the Finnish Loviisa VVER-440 plant [Paper III,V]. Three-dimensional APROS core models have been created for the Finnish Olkiluoto BWR plant [Paper VI] for transient and severe accident re-criticality analyses. Table 1 contains typical VVER core descriptions used in APROS for training simulator, plant analyser and safety analysis purposes. Table 2 gives corresponding information for some PWR and BWR core models.

The VVER1-VVER4 in Table 1 represent the VVER-440 type reactor core of the Finnish Loviisa plant. The core has 313 fuel assemblies and 37 control assemblies. In the APROS model, the neutronics part is described with 313 hexagonal fuel assemblies that have been divided into ten axial segments. In the models VVER1 and VVER2 of Table 1, six similar fuel assemblies were placed in the same thermal hydraulic channel. The central assembly was placed into its own channel. Thus, this alternative contained 53 channels that were described either using the five-equation or the six-equation thermal hydraulic model. These alternatives represent typical core sizes for training simulator use. The simplest way to represent asymmetric effects in a training simulator application in APROS is to describe the affected rod and the nearest neighbours with separate thermal hydraulic channels, which for the Loviisa core would mean seven additional channels, thus increasing the total number of thermal hydraulic channels from 53 to 60. In the models VVER3 and VVER4, each fuel assembly was placed in its own thermal hydraulic channel that was described either with the five- or the six-equation thermal hydraulic model. These core descriptions are suitable for engineering simulators and plant safety analyses.

The VVER5 in Table 1 is a VVER-1000 -type core of the advanced VVER-91 design (VVER-1000/91). The core contains 163 hexagonal fuel assemblies and 97 control rod clusters. In the APROS model, the core neutronics part has been described with 163 hexagonal fuel assemblies that have been divided into ten axial segments. The model was created for safety analyses and thus one fuel assembly per thermal hydraulic channel was used. Each thermal hydraulic channel was divided into ten sections and described with the five-equation thermal hydraulic model.

Table 1. Three-dimensional APROS VVER-type core models with various thermal hydraulic models and channel divisions.

Core type	**VVER1**	**VVER2**	**VVER3**	**VVER4**	**VVER5**
Fuel assemblies	313	313	313	313	163
Axial division	10	10	10	10	10
5-eq. Channels	53	-	313	-	163
6-eq. Channels	-	53	-	313	
Neutronics nodes	3 130	3 130	3 130	3 130	1 630
Thermal hydraulic nodes	530	530	3 130	3 130	1 630

The BWR1-BWR3 in Table 2 describe BWR-type core of the Finnish Olkiluoto plant. The core contains 500 fuel assemblies and 121 control assemblies. In the APROS model, the neutronics part was described with 500 quadratic fuel assemblies. Two alternatives were used in the description of thermal hydraulic flow channels: either two identical fuel assemblies were placed into each one-dimensional thermal hydraulic channel or four assemblies were placed into same thermal hydraulic channels. Thus, the total number of thermal hydraulic channels in the core was 250 or 125, respectively. Axially, each fuel assembly and thermal hydraulic channel was divided into 25 sections. In the BWR models, the thermal hydraulic channels were described either with the five-equation or the six-equation thermal hydraulic model. The extent of these core models represent the size required in safety analysis.

The PWR1 and PWR2 in Table 2 describe PWR-type core used for the calculation of the OECD NEA LWR core transient benchmark [45]. The core has 157 fuel assemblies that were divided into 16 axial segments. The model was created for the calculation of the benchmark, and in the model each fuel assembly was placed into its own thermal hydraulic channel. Both the five-equation and the six-equation models were used in the calculation of the transients.

Table 2. Three-dimensional APROS LWR-type core models with various thermal hydraulic models and channel divisions.

Core type	**BWR1**	**BWR2**	**BWR3**	**PWR1**	**PWR2**
Fuel assemblies	500	500	500	157	157
Axial division	25	25	25	16	16
5-eq. Channels	125	250	-	157	-
6-eq. Channels	-	-	250	-	157
Neutronics nodes	12 500	12 500	12 500	2 512	2 512
Thermal hydraulic nodes	3 125	6 250	6 250	2 512	2 512

4.2 Calculation speed considerations

In the early versions of APROS nuclear plant simulation models, first point kinetics and thereafter a one-dimensional two-energy group model with six delayed neutron groups was used for the description of the reactor core. The early applications aimed at real-time simulation of the plant, and thus the one-dimensional model was the most complicated one that could be used for core description with the VAX 8750 -type computers [37, 46].With the Alliant-FX/40 computer, larger models, like the three-dimensional core model, became possible. These applications with full three-dimensional neutronics with one or only a few thermal hydraulic channels in the core were still far beyond real time [Paper I].With the workstations like HP-9000/735, the true three-dimensional core models with an adequate number of thermal hydraulic channels, combined with the ability to calculate in real time have become possible [47]. The change of computer from VAX 8750 -type to HP-9000/735 and optimization of the code has increased the calculation speed of the same model by nearly a factor of fifty [48]. The performance of computers has increased at the same time according to the Linpack Benchmark [49] from the .99 Mflops for the DEC VAX 8700 and 2.4 Mflops for the Alliant FX/40 through 41 Mflops for HP-9000/735 to the 158 Mflops for HP C180-XP and 235 Mflops for the DEC AlphaStation 500/500 computers used at present at VTT Energy.

The calculation speeds reached with the core descriptions of Table 1 have been presented in Table 3. A time step size of 0.2 seconds is usual for training simulators at steady state. In safety analysis shorter time steps are used. It can be observed that for VVER-type reactors real-time calculation speed can at present be reached with cores where the symmetric fuel assemblies have been placed in the same five-equation thermal hydraulic channel. With cores consisting of a relatively small number of fuel assemblies, like the VVER-91, even the full core description with one fuel assembly per flow channel is already very close to the real time calculation speed.

Table 3. Calculation speeds with various VVER cores at steady state with various thermal hydraulic models. Time-step 0.2 s. Computer DEC AlphaStation 500/500. Speed indicated is simulation time/CPU time.

Core type	**VVER1**	**VVER2**	**VVER3**	**VVER4**	**VVER5**
Calculation speed	2.14	1.27	0.47	0.29	0.92

Table 4 shows the corresponding calculation times obtained with the core descriptions of Table 2. It can be noticed that the calculation speed obtained for the western PWR-core with the 5-equation thermal hydraulic model is quite fast. For the six-equation model the calculation time required is approximately double, as indicated also for the VVER-type cores. For the quite detailed BWR-type cores reaching of real-time calculation would still require reduction of axial division and lumping of fuel assemblies into macroelements and further placing identical macroelements into the same thermal hydraulic channel.

Table 4. Calculation speeds with various 3-D BWR and PWR cores at steady with various thermal hydraulic models. Time-step 0.2 s. Computer DEC AlphaStation 500/500. Speed indicated is simulation time/CPU time.

Core type	**BWR1**	**BWR2**	**BWR3**	**PWR1**	**PWR2**
Calculation speed	0.31	0.17	0.11	0.64	0.37

APROS reactor core models are designed to be used as the core models of plant analysers, training simulators, safety analysis or severe accident studies. In training simulators there is a strict real-time requirement for the entire model. In other applications, the calculation time requirement for a useful model is usually that an analysis must be performed overnight. Thus, the calculation time of the various plant process models must be measured, too. Since it has been observed [47] that the one-dimensional reactor model consumes approximately 2 percent of total calculation time in a plant analyser model, the various APROS plant analyser and training simulator applications having a one-dimensional core model can be used as a basis when estimating the time consumed by the plant process and automation description. Table 5 indicates the typical extent of some applications in terms of neutronics and thermal hydraulic nodes, pumps, valves, analog and binary signals and controllers. In Table 5, KOLA-AN is the Kola VVER-440 plant analyser [10], and KOLA-TR the corresponding compact training simulator [11], LOMO-6 is the Loviisa VVER-440 model used in recent safety analyses for Loviisa plant [50], and LOMO-5 is the corresponding model with 5-equation thermal hydraulics, and OLKI-5 and OLKI-6 are the Olkiluoto BWR plant analyser models [Paper VI].

Table 5. Number of components in various APROS BWR and VVER applications with one-dimensional core model and 5- or 6-equation thermal hydraulic model.

	KOLA-AN	**KOLA-TR**	**LOMO-5**	**LOMO-6**	**OLKI-5**	**OLKI-6**
Neutronics nodes	10	10	20	20	20	20
Thermal hydr. Nodes, 5-eq.	164	291	885	-	374	-
Thermal hydr. Nodes, 6-eq.	-	-	-	905	-	388
Thermal hydr. Nodes, 3-eq.	601	1173	123	103	193	193
Valves	244	835	206	206	158	158
Pumps	28	96	26	26	26	26
Analog signals	897	1 277	1 134	1 134	430	430
Binary signals	1 063	4 268	1 832	1 792	795	795
Controllers	31	54	32	32	6	6

The calculation speed of various applications with a one-dimensional core model have been given in Table 6. The difference in calculation speed between the Kola plant analyser and training simulator applications is due to the larger amount of thermal hydraulic nodes, pumps, valves, analog and binary signals in the training simulator, requiring a detailed description of the actual plant. The difference in the speed between the LOMO-5 and -6 models is due to the thermal hydraulic model selection. The OLKI-5 model is very fast, due to the relatively small number of thermal hydraulic nodes and automation components.

Table 6. Calculation speeds of various VVER and BWR applications with one-dimensional core model and 5- or 6-equation thermal hydraulic model. Time-step 0.2 s. Computer DEC AlphaStation 500/500. Speed indicated is simulation time/CPU time.

Simulator type	**KOLA-AN**	**KOLA-TR**	**LOMO-5**	**LOMO-6**	**OLKI-5**
Calculation speed	2.86	1.27	3.70	1.13	8.50

Studies performed for various three-dimensional APROS BWR and VVER type cores [Paper VIII] have indicated that the core thermal hydraulics required 58-90 % of the calculation time, and the neutronics solution required respectively only 10-42 % of the calculation time. The calculations performed with BWR, PWR and VVER-type cores indicated that the five-equation thermal hydraulic model was approximately twice as fast as the six-equation thermal hydraulic model per node. The difference in calculation speed originated mainly from the fact that with the five-equation model no iteration was performed, whereas with the six equation model at least one iteration round was included. The calculation speeds presented in Tables 3 and 4 for the three-dimensional cores, versus those given in Table 6 for the process and automation models, indicate that with the application of the parallel APROS concept [51] with the core model on one multiprocessor computer and the plant process and automation model on another multiprocessor computer, it would already be possible to construct real-time plant analysers and training simulators with three-dimensional core models.

5. Applications

5.1 Steady state comparisons and dynamics benchmarks

The one-dimensional core model of APROS has been validated using plant measurement data [52] and the HEXBU-3D code results [53]. Validation of the three-dimensional core model has included calculation of some international reactor dynamics benchmarks [44–45,54–56], comparison with plant measurement data [57] and comparison with the results of other codes [Paper VII].

The assembly powers of the three-dimensional APROS Loviisa core model have been compared with the measurement data of the Loviisa core in steady state. The agreement of the calculated and measured powers has been found to be quite good and the greatest relative deviations have been found in the low power assemblies at the outer rim of the reactor [57]. Figure 5a shows the relative assembly powers measured and those calculated by APROS for Loviisa unit 1 reactor core at beginning of cycle (BOC) 16 and Figure 5b shows the corresponding HEXBU-3D code results calculated at IVO and APROS code calculations. It can be noticed that the greatest relative deviations between APROS and measurement are found at the low power assemblies at the outer rim of the core. At the high power assemblies in the central part of core APROS tends to over-estimate the assembly power, as also observed in other comparisons [Paper VII]. The absolute deviations of the assembly powers calculated by APROS from the measured values vary from 4 % over-prediction for high power assembly in the vicinity of a control rod to 5 % under-prediction for low power assembly at the outer rim of the core. Figure 5b indicates rather good agreement between the values calculated with APROS and with HEXBU-3D code. The greatest relative deviations are again found at the low power assemblies at the outer rim of the reactor. The absolute deviations vary from 1 % over-prediction for some high power assemblies to 2 % under-prediction for low power assemblies at the outer rim of the core. In the APROS calculation in Figure 5a, the control rod group was assumed to be in the position indicated by plant data, and in the APROS calculation in Figure 5b the control rod group was assumed to be at the same position as in the HEXBU-3D calculation, which is the reason for the slightly different APROS assembly powers in the Figures.

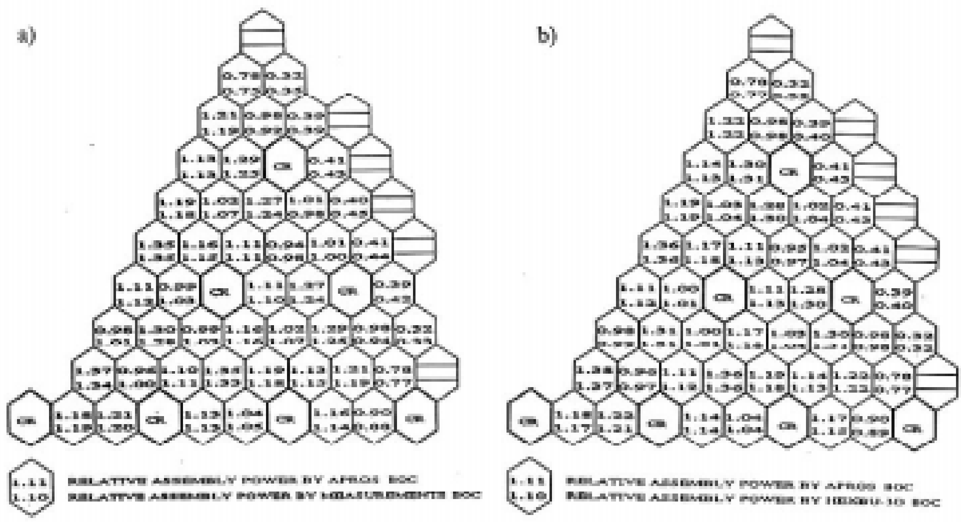

Figure 5. Comparison of APROS assembly powers with measured assembly powers (a) and with assembly powers calculated with HEXBU-3D code (b) for Loviisa 1, cycle 16, BOC.

The tendency of APROS to over-predict the power of high power assemblies was also noticed for the VVER-91 core, which has a larger node size. In this case the largest absolute over-prediction was found to be 11 % for the highest power assembly [Paper VII].

APROS has been used for re-criticality studies for a BWR core (see Chapter 5.5). The core loading used in the studies had half core symmetry. Thus two identical assemblies were placed in the same thermal hydraulic flow channel, which was described using the five- or six-equation thermal hydraulic model of APROS. In the re-criticality studies it was assumed that each flow channel had equal coolant inlet mass flow. In the studies the same extrapolation lengths were used in the core upper and lower boundaries and in the outer rim of the core with both thermal hydraulic alternatives. Figure 6 shows the calculated relative assembly powers obtained using either the five- or six-equation thermal hydraulic model in the flow channels and the relative assembly powers calculated by the plant fuel management code. It can be noticed that the largest absolute deviations of the APROS results from the plant fuel management code results vary from 7 % over-prediction for some high/medium power assemblies to 6 % under-prediction for low power assemblies at core outer rim. The axial profiles calculated by APROS have been presented together with the axial profile calculated by the plant fuel management code in Figure 7. It can be noticed that at the lower part of core the five-equation model underestimates the power peaking somewhat and the six-equation model predicts too strong axial power peaking. In the upper part of the core the situation is reversed.

.44 .44 .45	.38 .37 .38	.29 .26 .27										
.88 .94 .95	.66 .66 .67	.54 .55 .56	.43 .37 .37									
.97 .98 .99	1.11 1.13 1.14	.98 .97 .98	.68 .66 .67	.59 .55 .56	.45 .43 .44	.33 .29 .30						
1.13 1.14 1.15	1.23 1.24 1.25	1.14 1.14 1.14	1.11 1.09 1.10	1.01 1.03 1.04	.73 .71 .71	.58 .58 .59	.45 .40 .40					
.81 .78 .78	1.20 1.20 1.21	1.30 1.29 1.29	1.14 1.12 1.13	1.20 1.16 1.17	1.11 1.10 1.10	.99 .97 .98	.85 .88 .89	.52 .46 .46				
.80 .77 .77	1.13 1.14 1.14	1.30 1.27 1.27	1.34 1.31 1.32	1.10 1.04 1.04	1.20 1.15 1.16	1.02 1.00 1.00	1.06 1.04 1.05	.87 .91 .91	.47 .41 .42			
1.21 1.22 1.22	1.33 1.32 1.32	1.15 1.14 1.14	1.31 1.28 1.28	1.34 1.30 1.31	1.10 1.04 1.05	1.23 1.19 1.20	1.20 1.19 1.20	1.04 1.03 1.04	.62 .61 .61	.35 .31 .32		
1.06 1.06 1.05	1.32 1.31 1.31	1.33 1.33 1.33	1.14 1.15 1.15	1.25 1.25 1.25	1.34 1.33 1.33	1.18 1.16 1.16	1.26 1.24 1.24	1.17 1.18 1.19	.74 .77 .77	.48 .47 .47		
.90 .90	1.25 1.29	1.22 1.24	.83 .80	.85 .82	1.26 1.27	1.21 1.20	1.32 1.32	1.17 1.19	1.06 1.11	.62 .59		
.91 .92 .90	1.23 1.26 1.25	1.06 1.10 1.09	.83 .81 .80	.83 .81 .80	1.28 1.31 1.30	1.32 1.31 1.31	1.17 1.17 1.16	1.28 1.29 1.28	1.14 1.15 1.16	.73 .76 .76	.46 .41 .41	
1.24 1.28 1.27	1.22 1.25 1.24	1.32 1.35 1.34	1.20 1.23 1.22	1.13 1.17 1.16	1.33 1.32 1.32	1.37 1.38 1.38	1.12 1.08 1.08	1.25 1.25 1.25	1.22 1.24 1.24	1.02 1.03 1.03	.59 .59 .60	.32 .30 .30
1.22 1.28 1.27	1.29 1.32 1.31	1.23 1.27 1.26	1.08 1.11 1.10	1.21 1.24 1.23	1.15 1.16 1.16	1.31 1.31 1.31	1.08 1.09 1.09	1.22 1.25 1.25	1.22 1.22 1.23	1.11 1.13 1.14	.79 .84 .85	.41 .42 .42
.82 .81 .80	1.08 1.13 1.11	1.29 1.35 1.33	.92 .93 .91	.92 .93 .91	1.27 1.30 1.30	1.25 1.29 1.29	.82 .79 .79	.81 .78 .78	1.17 1.21 1.21	1.10 1.11 1.12	.88 .94 .95	.45 .46 .46
.82	1.22	1.24	.90	.90	1.06	1.21	.80	.81	1.13	.97	.88	.44

.81 .80	1.28 1.27	1.28 1.27	.91 .89	.90 .88	1.06 1.05	1.23 1.23	.77 .77	.79 .79	1.15 1.15	.99 .99	.95 .95	.45 .45
1.08 1.12 1.11	1.29 1.32 1.31	1.21 1.25 1.24	1.22 1.25 1.25	1.25 1.28 1.28	1.31 1.31 1.31	1.33 1.33 1.33	1.12 1.14 1.14	1.20 1.21 1.21	1.24 1.25 1.25	1.11 1.13 1.14	.66 .67 .67	.38 .37 .38
1.29 1.35 1.33	1.23 1.27 1.26	1.32 1.35 1.34	1.06 1.09 1.08	1.22 1.24 1.23	1.33 1.33 1.33	1.15 1.15 1.14	1.30 1.27 1.28	1.30 1.30 1.30	1.14 1.14 1.14	.98 .98 .99	.54 .55 .56	.29 .27 .27
.91 .92 .90	1.08 1.11 1.10	1.20 1.23 1.22	.83 .81 .80	.83 .80 .80	1.14 1.15 1.15	1.31 1.28 1.28	1.34 1.32 1.32	1.14 1.14 1.13	1.11 1.09 1.10	.68 .67 .67	.43 .37 .37	
.92 .93 .90	1.21 1.24 1.23	1.13 1.17 1.16	.83 .81 .80	.85 .81 .81	1.25 1.25 1.25	1.33 1.31 1.3§	1.10 1.05 1.05	1.20 1.17 1.17	1.01 1.04 1.04	.59 .56 .56		
1.27 1.30 1.29	1.15 1.16 1.15	1.32 1.32 1.32	1.28 1.30 1.30	1.26 1.26 1.26	1.34 1.32 1.32	1.09 1.04 1.04	1.20 1.15 1.16	1.11 1.10 1.11	.73 .71 .72	.45 .44 .44		
1.25 1.29 1.28	1.31 1.31 1.31	1.37 1.37 1.37	1.32 1.30 1.30	1.21 1.20 1.19	1.17 1.15 1.15	1.23 1.19 1.19	1.02 .99 1.00	.99 .97 .98	.58 .58 .59	.33 .30 .30		
.82 .78 .78	1.08 1.08 1.08	1.12 1.08 1.07	1.17 1.16 1.15	1.32 1.31 1.31	1.25 1.23 1.23	1.20 1.19 1.19	1.06 1.04 1.05	.85 .88 .89	.45 .40 .40			
.81 .78 .78	1.22 1.24 1.24	1.25 1.24 1.24	1.27 1.28 1.27	1.17 1.18 1.17	1.17 1.17 1.18	1.04 1.03 1.04	.87 .91 .91	.52 .46 .46				
1.17 1.20 1.20	1.22 1.21 1.22	1.22 1.23 1.23	1.14 1.14 1.15	1.06 1.10 1.10	.74 .77 .77	.62 .60 .60	.47 .41 .42					
1.10 1.11 1.12	1.11 1.12 1.13	1.01 1.02 1.02	.73 .75 .75	.62 .58 .59	.48 .47 .47	.35 .31 .32						
.88 .94 .94	.79 .84 .84	.59 .59 .59	.46 .40 .41									
.45 .45 .46	.41 .42 .42	.32 .30 .30										

.44	CALCULATION BY PLANT FUEL MANAGEMENT CODE (POLCA4)
.44	CALCULATION BY APROS 5-EQ. TH. MODEL
.45	CALCULATION BY APROS 6-EQ. TH. MODEL

Figure 6. Comparison of APROS assembly powers calculated using the five- and six-equation thermal hydraulic models in the flow channels and the asssembly powers calculated by the plant fuel management code for a BWR core.

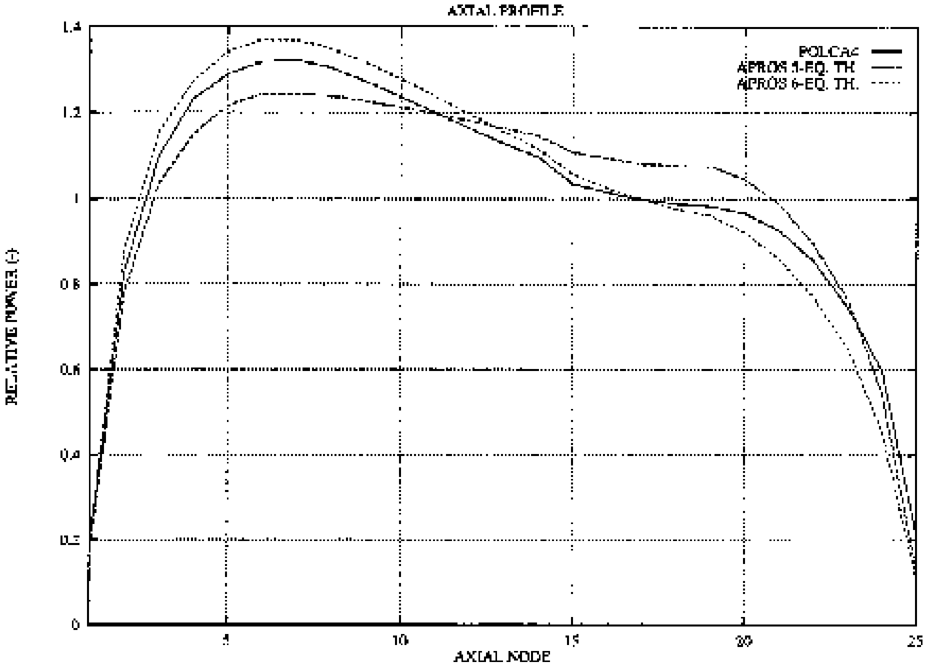

Figure 7. Comparison of APROS axial power profile calculated using the five- and six-equation thermal hydraulic models in the flow channels and the axial power profile calculated by the plant fuel management code (POLCA4) for a BWR core.

5.2 Plant analyser applications

Plant analysers were the first application area for APROS nuclear reactor core models. In the APROS environment, a plant analyser means a combination of models, user interface and graphics that are a prerequisite for performing safety analysis. In the early APROS plant analysers, a one-dimensional core model was used, and the emphasis was on the comparison of plant analyser performance versus training simulator data [58–62], or versus real plant data [62–66, Paper II]. In the real plant transients, reactor scram takes place relatively early in the course of the transient, and thus the emphasis is on the thermal hydraulic behaviour of the plant. In some of the real plant transients, the decisive factor for successful simulation has been the detailed description of the secondary side [Paper I] or description of operator actions [Paper III]. Due to the fact that reactor scram takes place quite early in these real plant transients the calculations with the three-dimensional core model have produced very similar results to the calculations with the one-dimensional core model. Another use of a plant analyser, often called an engineering simulator in this context, is to serve as a design tool when planning and performing plant design changes. APROS has been applied for these purposes, too, for VVER-440 [67] and VVER-1000/91 [68] plants.

The three-dimensional core model was first used in the context of the Loviisa plant analyser. The first studies were related to the effects of the amount of thermal hydraulic channels in symmetric core transients [32, 69]. The performance of the APROS 3-D core model of Loviisa VVER-440 plant was compared with plant data in a reactor trip and the model was used for the calculation of symmetric and asymmetric control rod withdrawal and ejection transients such as the ATWS cases [57].

The full three-dimensional core model of Loviisa plant has also been used for demonstrations of the capability of APROS to treat asymmetric control rod ejection or boron dilution transients [Paper IV]. In the example shown in Figure 8, one control assembly was ejected asymmetrically from the core when the reactor was in full power operation. Figure 8 shows the distorted power profile at one axial elevation at the upper part of the core at the time of maximum power of the total core and the time behaviour of total power and average fuel temperature in the core during five seconds of the transient.

Boron dilution transients can also be studied with the three-dimensional core model of APROS. In the example shown in Figure 9, the boric acid concentration at the core inlet of one sector of the core was suddenly decreased by 300 ppm [Paper IV]. Figure 9 shows the final distribution of boron concentration in the core, the power distortion in the core at the time of peak power and the behavior of core total power and average fuel temperature during ten seconds of the transient.

Examples of various symmetric and asymmetric core transients have also been given in Papers V,VI, VII and VIII and in reference [70].

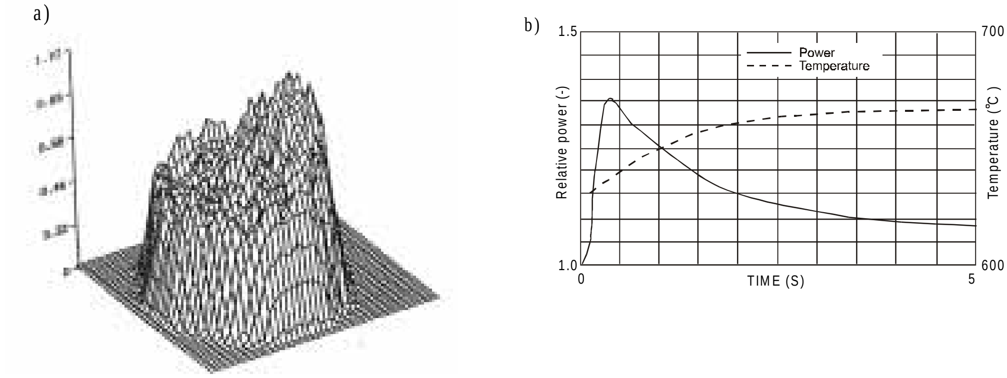

Figure 8. Asymmetric control assembly ejection. a) Power profile in relative units at core upper part at the time of maximum power. b) Total power in relative units and average fuel temperature in ° C during the transient. Adapted from [Paper IV].

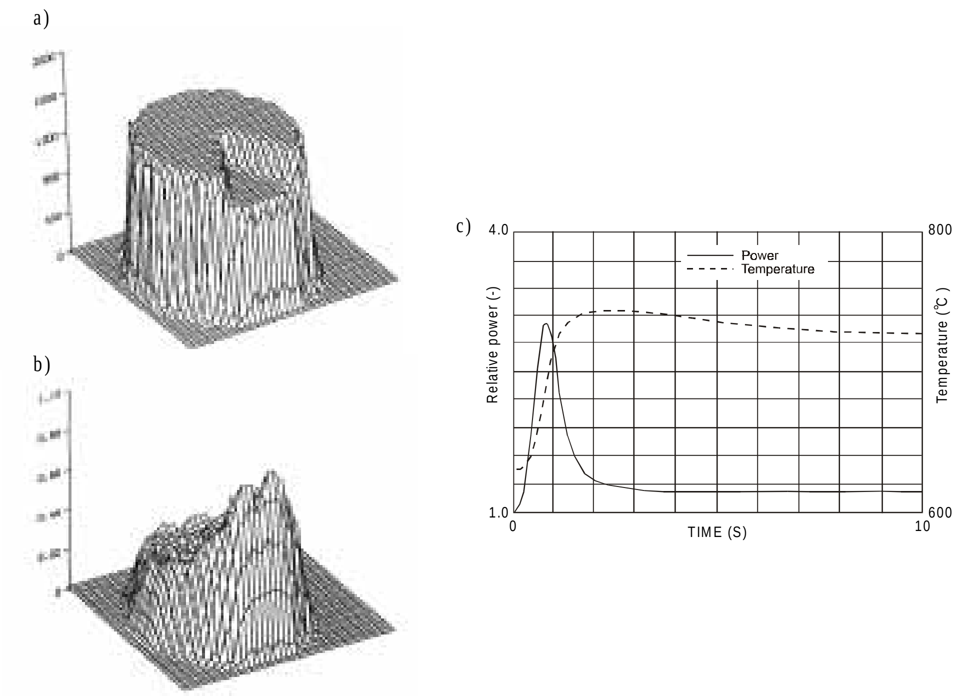

Figure 9. Asymmetric boron dilution transient. a) Boron distribution in core in ppm. b) Power distribution in core at time of peak power in relative units. c) Total power in relative units and average fuel temperature in ° C during the transient. Adapted from [Paper IV].

5.3 Safety analysis applications

APROS has been used extensively in plant safety analyses in the modernization and power uprating program for Loviisa Nuclear Power plant. In the years 1995-1997 the major part of FSAR safety analyses were recalculated [50]. The main tool in this work was the APROS code with the one-dimensional core model and six-equation thermal hydraulic model. The analyses performed with APROS included several large and small break LOCA (Loss Of Coolant Accident) analyses, ATWS (Anticipated Transient Without Scram) analyses comprising control rod withdrawal from full power, loss of main feed water and loss of on and off-site AC power. The analyses also included PRISE (PRImary-to-SEcondary-side-leakage) analyses, reactor coolant pump trips and seizure, main feed water pump trip, feed water line break, inadvertent closure of main steam line isolation valve, loss of on-site and off-site AC power, uncontrolled withdrawal of a control rod group during power operation, overpressure protection analysis, decrease of feed water temperature and inadvertent opening of one steam generator safety valve. In reference [50] an overview of these analyses has been given.

The use of three-dimensional core models is reasonable in the area of safety analysis, especially in asymmetric events in the core. In the safety analysis area, APROS three-dimensional core models have been applied in the calculation of various ATWS cases covering main steam line break, main steam header break and erroneous connection of a main reactor coolant pump for VVER-1000/91 type reactor [Paper VII].

For the VVER-1000/91, a detailed study of a main steam line break ATWS transient was performed with APROS and the results were compared with the results of HEXTRAN code in that transient [Paper VII]. The transient was calculated with both codes using full three-dimensional core description. The transient resulted in asymmetric core behaviour and the calculated results of the APROS and HEXTRAN codes were in good agreement with each other, which indicated that APROS was judged to be reliable enough for other similar ATWS transient analysis. The analysis work was continued with the calculation of the main steam header break and erroneous connection of main reactor coolant pump ATWS cases with APROS [Paper VII]. As an example of the results obtained Figure 10 shows the APROS and HEXTRAN results for calculated total power transferred to the coolant in the main steam line break transient.

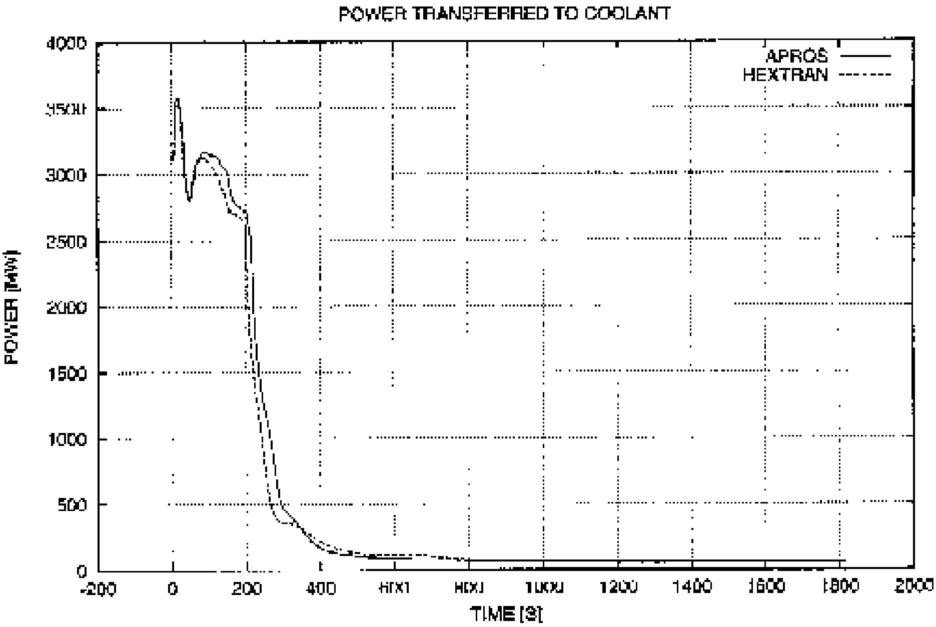

Figure 10. Power transferred to coolant in steam line break ATWS for VVER-91. APROS and HEXTRAN results.

5.4 Training simulator applications

In training simulators, the real time performance of the entire model is an essential requirement. For this reason the first APROS training simulator applications have used the one-dimensional core model. A compact training simulator containing the one-dimensional core model has been delivered to the Russian Kola plant [71]. The first application of APROS as the calculation routine in a full scope training simulator is the CHASNUPP training simulator project describing the Chinese 300 MWe PWR-type plant being built in Pakistan [14].

The running times reported in chapter 4 with the DEC AlphaStation 500/500 computer, together with those reported in reference [72] with a HP 180 C computer indicate that training simulator applications with three-dimensional core models are already a realistic alternative. In APROS, the three-dimensional core neutronics is always described with a finite difference type model and the thermal hydraulics with the five- or six-equation model. In a training simulator application, the faster five-equation thermal hydraulic model is always used. The required real-time or faster than real time calculation speed is already reached with some VVER-type core descriptions using the five-equation thermal hydraulics. In other applications, speedup can be obtained with lumping large parts of the core together both in neutronics and in thermal hydraulics. In training simulator applications, typically 10–15 axial nodes are used both in neutronics and in thermal hydraulics. In order to gain speed, identical fuel assemblies (usually 2, 4, 6 or 8 identical assemblies depending on reactor type and core loading) can be placed in the same thermal hydraulic channel. The fuel assemblies in radial direction can further be lumped into so-called macroelements, especially in BWR applications with a large number of fuel assemblies. The asymmetric effects in the core can be presented with a small group of fuel assemblies placed in separate flow channels (the concerned assembly and the closest 4–6 neighbours). A further way to ensure real time performance with the three-dimensional core is to use the parallel APROS application with the plant process and automation on one multiprocessor computer and the three-dimensional core model on another.

The first full scope training simulator using both one- and three-dimensional APROS core models is the HAMMLAB 2000 BWR training simulator being built for the OECD Halden Reactor Project in Norway [15]. The simulator concept is defined as an experimental simulator with three major application areas: performing experimental studies in the human factors programme, evaluating computerized operator support systems, and experimentation with advanced control room prototypes.

5.5 Severe accident applications

At VTT Energy, it is planned that the APROS programme should play a central role in the development and validation of calculation tools needed to plan preventive measures in severe accidents and in training the personnel for severe accident mitigation. Extension of APROS with severe accident models has just recently been started. As one of the first applications in the field of severe accidents, core re-criticality studies for BWR type reactors have been performed. The studies were initiated with a scoping study within the Nordic NKS-programme [12], and further studies are going on in an European Union project [13]. The objective of the re-criticality studies is to examine whether a BWR core can reach re-criticality in a severe accident when the core heat-up has caused a partial melting of the control rods and when the core is re-flooded with non-borated water from the emergency core cooling system. In the studies three computer codes are being used. Two of the codes, APROS and SIMULATE-3K, have their principal application area in core transient analysis, and the third, RECRIT [73], has been specially developed for re-criticality studies.

Codes like APROS or SIMULATE-3K can describe the core in more geometric details than the actual severe accident codes, like MAAP [74] or MELCOR [75]. The severe accident codes describe only the decay heat effect. Thus, a code containing a core neutronics model, like APROS, SIMULATE-3K or RECRIT, has to be used to study the possibility of core re-criticality. Other benefits of codes like APROS are a more advanced thermal hydraulics description, the possibility to describe asymmetric events in the core, like asymmetric re-flooding or control rod melt in the core, the possibility to use true assembly-based core description, and to combine the core with process and containment models. With codes like APROS it can still be considered reasonable to discuss the situation at the initiation of re-criticality. However, the validity range of APROS or SIMULATE-3K for re-criticality studies is limited at present to the period when the core materials still have reasonable temperatures and it can be assumed that fuel rods and assemblies preserve their original position and geometry. Thus, the time window where the use of this kind of a code is reasonable is quite limited. Discussion of the progressing features of severe accidents requires development of specific models. Such models are being developed in separate projects beyond this study.

The initial conditions of APROS re-criticality studies were determined with the severe accident codes MAAP and MELCOR. In the example, the APROS core was brought to the same fuel temperatures as indicated by MELCOR, 65 % of control rods were assumed to be melted and the effects of four re-flooding rates, also based on the MAAP and MELCOR analysis, were studied. In the studies the three-dimensional core model of the Finnish Olkiluoto plant of the TVO utility was used. The core model contains 500 fuel assemblies and 250 thermal hydraulic channels [Paper VI]. The re-flooding studies were performed using both the five- and the six-equation thermal hydraulic models in the flow channels. The results for power peak size were very similar with both thermal hydraulic models, although there were some differences in the predicted fuel temperatures and void fractions. Figure 11 shows the resulting power peaks with two different re-flooding rates, and Figure 12 illustrates the neutron flux, fuel temperature and void fraction distribution at one axial section at the lower part of core, at the axial section with maximum power and at one axial section at the upper part of the core at the time of power peak.

The re-criticality studies are continuing with the combination of the three-dimensional core model and plant process model to determine the consequences of the re-criticality peaks with a larger time frame in a more realistic manner. The continued studies will also contain inclusion of the specially developed severe accident models.

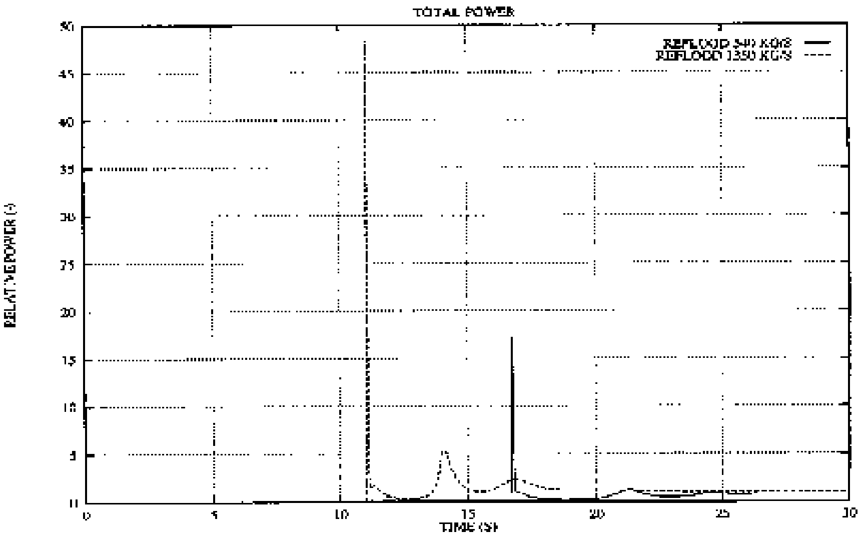

Figure 11. Power peak obtained at various reflooding rates in a BWR core. Power relative to full nominal power.

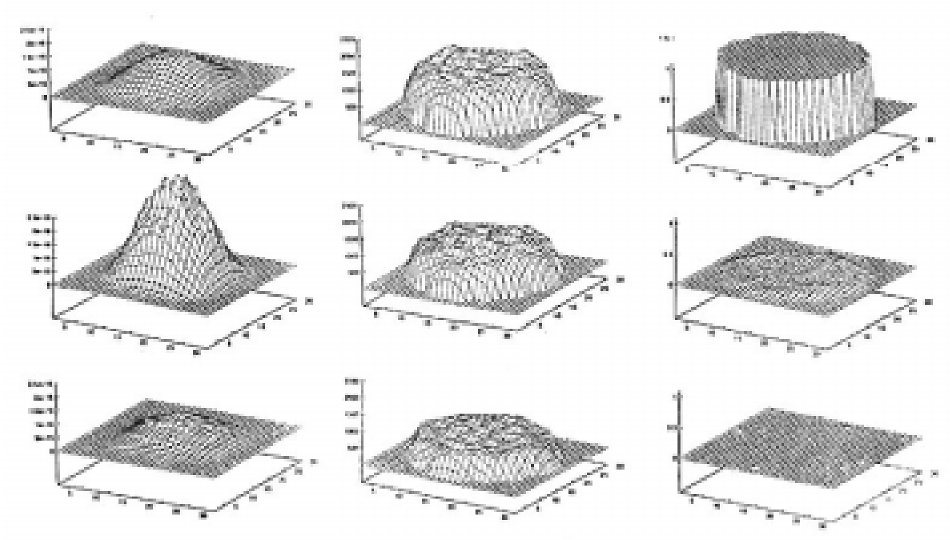

Figure 12. Distribution of fast neutron flux, fuel temperature and void fraction at various axial elevations of a BWR core during re-criticality power peak.

6. Conclusions

This thesis describes the author´s work in developing, validating and applying the nuclear reactor core models of the APROS multifunctional simulation environment. The APROS environment contains one- and three-dimensional nuclear reactor models, both created by the author. At present, APROS applications in nuclear area cover plant analysers or engineering simulators, plant safety analysis, training simulator applications and also some severe accident studies. There are different requirements set for the nuclear reactor core model in these various areas, and also within each area there are varying requirements, depending on the situation analysed.

The APROS environment is unique in the sense that the same detailed physical models are being used in all application areas extending from training simulators to severe accident analysis, which enhances the reliability of the models. The existence of two reactor models, one- and three-dimensional, as well as the existence of the alternative five- and six-equation thermal hydraulic models, is considered as a strength of the calculation system. Each model combination clearly has its proper area of application.

The emphasis in this thesis has been on the three-dimensional core model and its capability to analyses symmetric and asymmetric events in the core. The core model results have been compared with the Loviisa plant measurement data in steady state and in some transients. Three-dimensional core models have been created for the calculation of international benchmarks for BWR, PWR and VVER-type cores. Various three-dimensional core models have also been created for calculation of control rod ejection and boron dilution transients for the Loviisa VVER-440 core and for the severe accident studies using Olkiluoto BWR-core model. The most demanding task of the APROS three-dimensional core model so far has been the calculation of several main steam line break type ATWS analyses for the VVER-1000/91 reactor. In this context, the results of APROS were also compared in detail with the results of the HEXTRAN code, and APROS was judged to be reliable enough for such analyses. Comparisons of APROS three-dimensional core model results with the plant measurement data and plant fuel management code results show that the APROS three-dimensional core model is accurate enough for plant analyser and training simulator applications. However, the accuracy may not be good enough for all safety analysis applications.

The one-dimensional core model has been used extensively in the past in calculating various thermal hydraulic plant transients. It was also used extensively in the modernisation and power uprating process of the Loviisa plant. The one-dimensional model will have its proper application areas in future, too, in cases where ultimate speed is required and where events in respect of reactor core are symmetric. However, the development of computer calculation capacity already allows calculation of all safety analyses with asymmetric core response with the three-dimensional core model. The calculation times of three-dimensional core models presented in the thesis indicate that three-dimensional core modelling is also becoming a realistic alternative in training simulators.

References

1. Silvennoinen, E., Juslin, K., Hänninen, M., Tiihonen, O., Kurki, J. & Porkholm, K. The APROS Software for Process Simulation and Model Development. Espoo: Technical Research Centre of Finland, 1989. 106 p. + app. 19 p. (Research Reports 618.) ISBN 951-38-3463-8.

2. Tiihonen, O. & Juslin, K. “Requirements for a Multifunctional Code Architecture”. Proceedings of the OECD/CSNI Workshop on Transient Thermal-Hydraulic and Neutronic Codes Requirements. Annapolis, Md, USA, 5–8 November 1996. U.S. Nuclear Regulatory Commission, 1997. NUREG/CP-0159. Pp. 716–723.

3. Juslin, K. & Paljakka M. APROS - A Multifunctional Modelling Environment. Paper presented at the 2nd CSNI Specialist Meeting on Simulators and Plant Analysers, Hanasaari, Espoo, Finland 29 September–2 October 1997. 10 p.

4. Ransom, V. H. et al. RELAP5/MOD2 Code Manual. EG & G Idaho, Inc. Vol. 1 & 2, 1985. (NUREG/CR-4312.)

5. Micaelli, J. C. A Description of the CATHARE Code. Proceedings of Lappeenranta Seventh Summer School on Heat Transfer, Lappeenranta, Finland 6–10 August 1990. Lappeenranta: Lappeenranta University of Technology. (Publication EN D-22.) ISBN 951-763-650-4. 17 p.

6. Kyrki-Rajamäki, R. Three-Dimensional Reactor Dynamics Code for VVER Type Nuclear Reactors. Espoo: Technical Research Centre of Finland, 1995. 51 p. + app. 80 p. (VTT Publications 246.)

7. Cronin, T. J., Smith, K. S. & Ver Planck, D. M. SIMULATE-3 Methodology, 1995 (Studsvik/SOA-95/18.)

8. Kaloinen, E. & Kyrki-Rajamäki, R. TRAB-3D, A New Code for Three-dimensional Reactor Dynamics, CD-ROM Proceedings of ICONE-5, 5th International Conference on Nuclear Engineering. Nice, France 26–30 May 1997. Paper ICONE5-2197. 8 p.

9. Porkholm, K., Nurmilaukas, P., Tiihonen, O., Hänninen, M. & Puska, E. K. Loviisa Nuclear Power Plant Analyser. Helsinki: Imatran Voima, 1992. 48 p. + app. 22 p. (Research Reports IVO-A-11/92.) ISBN 951-8928-92-4.

10. Porkholm, K., Hänninen, M., Shutov, V. & Ovtcharova, I. APROS-Based Kola Nuclear Plant Analyser. Proceedings of the 1994 SCS Simulation Multiconference. La Jolla, San Diego, CA 10–13 April 1994. The Society for Computer Simulation. Jaime Olmos & Ariel Sharon (eds.). XI Annual Simulators Conference (Simulation Series) Vol. 26. No 3. Pp. 93–98.

11. Porkholm, K., Kontio, H., Nurmilaukas, P., Tiihonen, O. Hänninen, M., Ylijoki, J., Kumkov, L., Netchaev, S. & Pirog, V. APROS-Based Kola 1 Nuclear Power Plant Compact Training Simulator. Proceedings of the 1996 Simulation Multiconfrence. New Orleans, Louisiana l 8–11 April 1996. The Society for Computer Simulation SCS. Ades, Maurice & Griebenov, Ron (eds.). Simulators International XIII. (Simulation Series) Vol. 28. No 2. Pp. 3–8.

12. Hojerup, F., Miettinen, J., Nilsson, L., Puska, E. K., Sjövall, H., Anttila, M. & Lindholm, I. On Recriticality During Reflooding of a Degraded Boiling Water Reactor Core. Nordic Nuclear Safety Research NKS 1997. (NKS/RAK-2(97)TR-A3.) ISBN 87-7893-010-3.

13. Frid, W., Nilsson, L., Puska, E. K., Höjerup, F., Miettinen, J., Lindholm, I. & Sjövall, H. Severe Accident Recriticality Analysis. Proceedings of the FISA´97 Meeting, Luxembourg, November 1997. European Commission. Van Goethem, G. et al. (eds.). Report EUR 18258 EN. Pp. 133–142.

14. Hassan, M. W., Khan, N. M., Ali, S. & Jafri, M. N. Verification and Validation Methodology of Training Simulators. Paper presented at the IAEA Specialists Meeting on “Training Simulators in NPPs”. Experience Programme Design and Assessment Methodology. Essen, Germany 17–19 November 1997. 13 p.

15. Tiihonen, O. and Puska, E. K., APROS-Based BWR Simulator for HAMMLAB 2000. Paper presented at the Enlarged Halden Programme Group Meeting, March 15–20, 1998, Lillehammer, Norway.

16. Meinander, M. Process Component Definition in the Simulation of Thermal Hydraulic Processes. Helsinki University of Technology, Department of Engineering Physics and Mathematics, 1997. Master´s Thesis. 63 p. + app. 8 p.

17. Kyrki-Rajamäki, R. VTT Energy´s Calculation System for Reactor Physics and Dynamics. Proceedings of the 7th Nordic Reactor Physics Meeting "Reactor Physics Calculations in Nordic Countries", Espoo 8–9 May 1995. Espoo: Technical Research Centre of Finland (VTT Symposium 159.) Pp. 67–78.

18. Rajamäki, M. TRAWA, A Transient Analysis Code for Water Reactors. Espoo: Technical Research Centre of Finland, Nuclear Engineering Laboratory, 1976. 149 p. + app. 31 p. (Report 24.) ISBN 951-33-0352.

19. Rajamäki, M. TRAB, a Transient Analysis Program for BWR. Part 1. Principles. Helsinki: Technical Research Centre of Finland, Nuclear Engineering Laboratory, 1980. 101 p. + app. 9 p. (Report 45.)

20. Höglund, R. A Nodal Method for Calculating Power Distributions for LWR-Type Reactors with Square Fuel Lattices. Helsinki: Technical Research Centre of Finland, Nuclear Engineering Laboratory, 1980. 27 p. (Research Report 51.) ISBN 951-38-1012-7.

21. Kaloinen, E., Teräsvirta, R. & Siltanen, P. HEXBU-3D, a Three-Dimensional PWR-Simulator Program for Hexagonal Fuel Assemblies. Espoo: Technical Research Centre of Finland, 1981. 148 p. + app. 5 p. (Research Reports 7/1981.) ISBN 951-38-1254-5.

22. Kyrki-Rajamäki, R. Application of Coupled 3D Neutronics and Circuit Thermal Hydraulics in Reactor Analysis. Proceedings of the Joint International Conference on Mathematical Methods and Supercomputing for Nuclear Applications (M&C+SNA´97). Saratoga Springs, NY 5–10 October 1997. La Grange Park,IL: American Nuclear Society, 1997. Vol. 2. Pp. 1349–1357. ISBN 0-89448-619-5.

23. Judd, J., Downar, T., Turinsky, P., Barber, D., Joo, H., Sarsour, H., Ross, K., Clement, S., Wagner, K. C., Wagner, R. J. & Beelman, R. J. High-Fidelity,

Real-Time Simulation with RELAP5/NESTLE. Trans. Am. Nuc. Soc., 1996. Vol. 75, p. 321.

24. Hagrman, D. T. & St. Clair, R. SIMULATE-3K RETRAN-3D Coupled Systems Calculations. Advances in Nuclear Fuel Management II. Myrtle Beach, March 20–21, 1997.

25. Kliem, S., Grundmann, U. & Rohde, U. The Coupled Code Complex DYN3D/ATHLET - Application to Main Steam Line Break Analysis. Proceedings of the Joint International Conference on Mathematical Methods and Supercomputing for Nuclear Applications (M&C+SNA´97) Saratoga Springs, NY 5–10 October 1997. La Grange Park, IL: American Nuclear Society, 1997. Vol. 2. Pp. 1358–1366. ISBN 0-89448-619-5.

26. Feltus, M. A. & Labowski, K. PWR Anticipated Transients Without Scram Analyses Using PVM- Coupled RETRAN and STAR Three-Dimensional Kinetics Codes. Proceedings of the Joint International Conference on Mathematical Methods and Supercomputing for Nuclear Applications (M&C+SNA´97). Saratoga Springs, NY 5–10 October 1997. La Grange Park, IL: American Nuclear Society 1997, Vol. 2. Pp. 1367–1379. ISBN 0-89448-619-5.

27. Aragones, J. M., Ahnert, C. & Aragones, V. Coupled 3-D Neutronic-Thermalhydraulic Analysis of Transients in PWR Cores. Proceedings of the Joint International Conference on Mathematical Methods and Supercomputing for Nuclear Applications (M&C+SNA´97). Saratoga Springs, NY 5–10 October 1997. La Grange Prk, IL: American Nuclear Society, 1997. Vol. 2. Pp. 1380–1390. ISBN 0-89448-619-5.

28. Rajamäki, M. & Saarinen, M. PLIM Shape-preserving Characteristics Method for Flow Equations. Proceedings of International Topical Meeting on Advances in Mathematics, Computations, and Reactor Physics. Pittsburgh, PA 28 April–1 May 1991 La Grange Park, IL: American Nuclear Society, 1991. Vol. 3. Pp. 12.1 1-1–12.1 1-13.

29. Kyrki-Rajamäki, R. & Räty, H. Neutronics and Thermal Hydraulics Modelling in Reactor Dynamics Codes TRAB-3D and HEXTRAN, Paper

presented at the 2nd CSNI Specialist Meeting on Simulators and Plant Analysers. Hanasaari, Espoo: 29 September–2 October 1997.

30. Jakobsen, O., Johnsen, T. & Randem, H. The Picasso-3 User Interface Management System. Paper presented at the Enlarged Halden Programme Group Meeting, Storefell, Norway, March 8–11, 1993. Halden: OECD Halden Reactor Project.

31. Williams, T. & Kelley, C. GNUPLOT, http://www.cs.darthmouth.edu/gnuplot/gnuplot.html.

32. Puska, E. K., Niemi, J. & Kontio, H. Reactor Core Calculations in Plant Analyser Environment, Proceedings of M&C+SNA'93. Karlsruhe, Germany 19–24 April 1993. Karlsruhe: Kernforshungszentrum Karlsruhe GmbH. Küsters, H. et al. (eds.). 1993. Vol. 1. Pp. 818–830.

33. Puska, E. K., Hämäläinen, A., Kontio, H. & Niemi, J. Interactive Simulation and Visualisation in APROS Nuclear Applications. Proceedings of the SIMS'94 Simulation Conference. Stockholm, Sweden 17–20 August 1994. Computer Solutions Europe AB and SIMS. Langemyr, L. (ed.). Pp. 150–155.

34. Puska, E. K., Hämäläinen, A. & Kontio, H. Visualisation in Large Plant Analyser Applications. Proceedings of the 1995 Simulation Multiconference, High Performance Computing Symposium. Phoenix, Arizona 9–13 April 1995. The Society for Computer Simulation. Pp. 429–434. ISBN 1-56555-078-1.

35. Puska, E. K. & Kontio, H. APROS Simulation Environment in Nuclear Plant Simulation. 1995 Summer Computer Simulation Conference. Ottawa, Ontario 24–26 July 1995. The Society for Computer Simulation. Pp. 703–708. ISBN 1-56555-081-1.

36. Puska, E. K. One- and Three-Dimensional Nuclear Reactor Core Models for Modular Plant Analyser. International Topical Meeting on the Safety of Thermal Reactors. Portland, USA 21–25 July 1991. American Nuclear Society. Pp. 253–260.

37. Juslin, K., Hänninen. M., Puska, E. K., Kurki, J., Nyström, P. & Porkholm, K. Application of an Advanced Process Simulator to Nuclear Engineering. Specialists' Meeting on Training Simulators for Nuclear Power Plants. Toronto, Canada 14–16 September 1987. Vienna: International Atomic Energy Agency IAEA. 13 p.

38. Ahlin, Å, Edenius, M. & Häggblom, H. CASMO, A Cell and Assembly Spectrun Code. Nyköping, Sweden: Studsvik Energiteknik, 1980. 11 p. + app. 11 p. (Ny Studsvik/RF-77/6276).

39. CASMO-HEX is based on the CASMO Code of Swedish origin. In: Ahlin, Å, Edenius, M. And Häggblom, H., CASMO, A Cell and Assembly Spectrum Code. Nyköping, Sweden: Studsvik Energiteknik, 1980. 11 p. + app. 11 p. (Ny Studsvik/RF-77/6276).

40. Peltonen, J. High and Second Order Polynomial Feedback Models for VVER-440 Fuel Two-Group Cross Sections. Proceedings of the 6th Symposium of AER. Kirkkonummi, Finland 23–26 September 1996. Budapest: AER 1996. Pp. 333–343. ISBN-963-372-607-7.

41. Hänninen, M. Thermohydraulic Models of APROS. In: Juslin, K. & Silvennoinen, E. (eds.). Numerical simulation of processes. Espoo: Technical Research Centre of Finland, 1988. Pp. 22–33. (VTT Symposium 84.) ISBN 951-38-3119-1.

42. Miettinen, J. & Hänninen, M. APROS Five Equation Thermohydraulics AFET. Espoo: VTT Nuclear Engineering Laboratory, 1993. 26 p. (Technical Report HOHTI-5/93).

43. Hänninen, M. & Ylijoki, J. The One-Dimensional Two-Fluid Model of APROS, Espoo: VTT Nuclear Engineering Laboratory, 1992. 75 p. (Technical Report HOHTI-2/92).

44. Kontio, H. Kolmedimensioisen reaktorimallin termohydrauliikka. Thermal Hydraulics of Three-Dimensional Nuclear Reactor Core Model. Espoo: Helsinki University of Technology, Faculty of Mechanical Engineering, 1993. Master´s Thesis. (In Finnish)

45. Leppänen, A.-P. Calculation of NEANSC 3D LWR Core Transient Benchmark PWR Full Power cases with APROS Version 4.05. Espoo: VTT Energy, Nuclear Energy, 1997. 22 p. (Technical Report ROIMA-9/97.)

46. Hänninen. M., Puska, E. K. & Nyström, P. Design and Analysis of Nuclear Processses with the APROS. Proceedings of the International Nuclear Simulation Symposium and Mathematical Modelling Workshop. Schliersee, Germany October 1987. Heidelberg, Berlin: Springer Verlag, 1987. Pp. 188–204, ISBN-0-387-18347-7.

47. Puska, E. K. & Kontio, H. Real-Time Simulation of Nuclear Power Plants with APROS. Proceedings of 1994 SCS Simulation Multiconference, High Performance Computing Symposium. La Jolla, CA 10–15 April 1994. The Society for Computer Simulation. Pp. 194–199. ISBN 1-56555-073-0.

48. Tiihonen, O., Hänninen, M., Puska, E. K., Porkholm, K. & Nurmilaukas, P. APROS-Based Loviisa Nuclear Power Plant Analyser. Proceedings of the OECD/CSNI Specialist Meeting on Simulators and Plant Analysers. Lappeenranta, Finland 9–12 June 1992. (NEA/CSNI/R(93)11.) Paris: OECD, 1994. Pp. 510–525, ISBN 951-38-4092-1.

49. Dongarra,J., LinpackBenchmark, http://performance.netlib.org/performance/html/linpack.data.co10.html, 1998.

50. Kantee, H., Kallio, H. & Savolainen, S. Experience of APROS in Nucler Power Safety Analyses. The Second CSNI Specialist Meeting on Simulators and Plant Analysers. Hanasaari, Espoo, Finland, 29 September–2 October 1997. 14 p.

51. Tiihonen, O., Piponius, T. & Porkholm, K. PARAS (Parallel Design and Analysis Software), CSC News, Vol. 10, No 1, March 1998. Pp. 10–13.

52. Puska, E. K., Hänninen, M., Ylijoki, J., Porkholm, K., Nurmilaukas, P. & Kontio, H. Summary of APROS Loviisa Plant Analyser Model Validation Tests I. Espoo: VTT Energy, 1993. (Technical Report HOHTI-6/93.)

53. Kuusisto, J. & Antila, M. Description and Validation of the 1-Dimensional Core Model for the APROS Analyser of Loviisa NPP. Proceedings of the 6th Symposium of AER. Kirkkonummi, Finland, 23–26 September 1996. Budapest: AER, 1996. Pp. 77–85. ISBN-963-372-607-7.

54. Puska, E. K. & Kontio, H. APROS 3-D results for AER Benchmark 1A. Proceedings of the third Symposium of AER. Piestany, Slovakia 27 September–1 October 1994. Budapest; AER. Pp. 283–291.

55. Puska, E. K. & Kontio, H. APROS 3-D Results for the NEANSC 3D LWR Core Transient Benchmark (3DLWRCT) in June 1993. Espoo: VTT Energy, Nuclear Energy, 1998. 100 p. (Technical Report ROIMA-3/98.)

56. Puska, E. K. & Kontio, H. APROS 3-D Results for the NEANSC 3D LWR Core Transient Benchmark (3DLWRCT) Case E2. Espoo: VTT Energy, Nuclear Energy, 1993. 20 p. + app. 10 p. (Technical Report HOHTI-4/93.)

57. Puska, E. K. & Kontio, H., Transient and Accident Analysis Capability of Plant Analyser with Three-Dimensional Core Model. Proceedings of ANS 1994 Topical Meeting on Advances in Reactor Physics. Knoxville, Tennessee 11–15 April 1994. La Grange Park: American Nuclear Society. Vol. III. Pp. 123–132.

58. Tiihonen, O., Puska, E. K., Hänninen, M., Virtanen, J. & Kurki, J. Advanced Simulator for Reactor Safety Analysis. Proceedings of International ENS/ANS Conference on Thermal Reactor Safety, Avignon, October 1988. Pp. 40-1–40-10.

59. Puska, E. K., Hänninen, M., Miettinen, J., Virtanen, J., Kurki, J. & Linden, U. Simulation of Loviisa Nuclear Power Plant Transients with APROS. Proceedings of the Topical Meeting on Advances in Nuclear Engineering Computation & Radiation Shielding. Santa Fe, New Mexico 9–13 April 1989. La Grange Park, IL: American Nuclear Society, 1989. Pp. 28:1–28:12.

60. Puska, E. K., Hänninen, M., Saarinen, M. & Linden, U. Simulation of ATWS Transients with APROS. Proceedings of the International Conference on the Physics of Reactors: Operation, design and computation, PHYSOR-90. Marseille 23–27 April 1990. Vol. 4. Pp. PV-95–PV105.

61. Puska, E. K., Saarinen, M. & Porkholm, K. APROS Nuclear Plant Analyser. Proceedings of the Second European Nuclear Simulation Symposium. Schliersee, Germany, October 1990, Berlin, Heidelberg: Springer Verlag, 1990. Pp. 193–204. ISBN-0-387-53085-1.

62. Puska, E. K. Nuclear Power Plant Analyser. 7th Summer School on Heat Transfer. Lappeenranta, Finland, 6–10 August 1990. Lappeenranta: Lappeenranta University of Technology, Department of Energy Technology. 16 p. + app. 2 p. (EN D-22, vol. III.) ISBN 951-763-651-2.

63. Puska, E. K.& Porkholm, K. Plant Analyser of Loviisa NPP. Proceedings of the International Topical Meeting on Advances in Mathematics, Computations, and Reactor Physics. Pittsburgh, PA, 28 April–1 May 1991. La Grange Park, IL: American Nuclear Society. Pp. 12.2 4-1–12.2 4-12.

64. Porkholm, K., Puska, E. K., Tiihonen, O. & Ylijoki, J. Simulation of Transients with Loviisa Plant Analyser. SIMS'92: Simulation Conference on Simulation of Chemical and Power Plant Processes and Annual meeting of Scandinavian Simulation Society. Lappeenranta, Finland, 10–12 June 1992. Lappeenranta: Lappeenranta University of Technology, Department of Energy Technology, 1992. P. 14. (EN D-26.) ISBN 951-763-729-2.

65. Porkholm, K., Hänninen, M., Puska, E. K. & Ylijoki, J. APROS Code for the Analysis of NPP Thermal Hydraulic Transients. Proceedings of the Nuclear Thermal Hydraulics. ANS Winter Meeting. Chicago, IL, 15–20 November 1992. La Grange Park, IL: American Nuclear Society, 1992. Pp. 17–25.

66. Hänninen, M., Porkholm, K., Puska, E. K. & Ylijoki, J. Analysis of Loviisa NPP Thermal Hydraulic Transients with APROS. Proceedings of the Third International Symposium on Power plant transients. 1992 ASME Winter Annual Meeting. November 8–13 November 1992, Anaheim, CA. New York, NY: The American Society of Mechanical Engineers, Power Plant Transients. (FED-Vol. 140.) Pp. 135–145.

67. Tiihonen, O., Porkholm, K., Hänninen, M. & Linden, U. Experiences of the Loviisa Plant Analyser Based on the APROS Simulation Environment. Proceedings of the IAEA Specialists´ Meeting on Training Simulators for Safe

Operation in Nuclear Power Plants. Balatonfured, Hungary, 24–27 September 1991. Vienna: International Atomic Energy Agency, 1992. Pp. 346–355.

68. Nurmilaukas, P., Porkholm, K. & Kontio, H. The APROS Simulation Model for VVER-91 Power Plant. Proceedings of SIMS´92 Simulation Conference. Lappeenranta, Finland, 10–12 June 1992. Lappeenranta: Lappeenranta University of Technology (LTKK EN D-26.) 8 p. ISBN 951-763-729-2.

69. Puska, E. K., Hämäläinen, A., Tiihonen, O., Porkholm, K. & Kontio, H. Three-Dimensional Core Neutronics Simulation in APROS. Proceedings of Third European Nuclear Simulation Symposium. Schliersee, Germany October 1993. The Society for Computer Simulation, 1993. Pp. 87–98. ISBN 1-56555-058-7.

70. Juslin, K., Miettinen, J., Puska, E. K., Porkholm, K. & Kontio, H. Analysis of VVER-440 Safety, Performance of APROS Simulation Environment in RIA, ATWS and LOCA Cases. Proceedings of CISS - First Joint Conference of International Simulation Societies, Zurich, Switzerland 22–25 August 1994. Pp. 216–220. ISBN 1-56555-031-5.

71. Porkholm, K., Honkoila, K., Nurmilaukas, P. & Kontio, H. APROS multi-functional simulator for thermal and nuclear power plants. Proceedings of the 1st World Congress on Systems Simulation (WCSS 97), 1–3 September 1997, Singapore. Society for Computer Simulation (SCS). Teo, Y. M. et al. (eds.). Pp. 504–508. SCS-ISBN 1-56555-114-1.

72. Puska, E. K. APROS 3-D Core Models for Simulators and Plant Analysers. The Second CSNI Specialist Meeting on Simulators and Plant Analysers. Hanasaari, Espoo, Finland. 29 September–2 October 1997.

73. Höjerup, F. A Simple Model for Assessing the Possibility and the Consequences of Recriticality during the Reflooding of a Damaged Core. Roskilde, Danmark: Riso National Laboratory, 1995. (Report Riso-I-909.)

74. MAAP4 - Modular Accident Analysis Program for LWR Power Plant - Computer Code Manual, 1–4, EPRI Research Project 3131-02, Fauske & Associates, Inc., Burr Ridge, IL, May 1994.

75. Summers, R. M., Cole, R. K. Jr., Smith, R. C., Stuart, D. S., Thompson, S. L., Hodge, S. A., Hyman, C. R. & Sanders, R. L. MELCOR Computer Code Manuals. Sandia National Laboratories, 1995. Vols. 1 & 2. 1 680 p. (NUREG/CR-6119, SAND93-2185.)

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